UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

August 10, 2006

Tennessee Valley Authority
ATTN: Mr. K. W. Singer
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INTEGRATED INSPECTION REPORT 05000259/2006007

Dear Mr. Singer:

On July 15, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a quarterly inspection period associated with recovery activities at your Browns Ferry 1 reactor facility. The enclosed integrated inspection report documents the inspection results, which were discussed on August 1, 2006, with Mr. Masoud Bajestani and other members of your staff.

We previously informed you, in a letter dated December 29, 2004, of the transition of four Reactor Oversight Process (ROP) Cornerstones (Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection) to be monitored under the ROP baseline inspection program. Consequently, as of January 2005, inspections for these cornerstones are integrated with Unit 2 and 3 ROP baseline inspections and Integrated Quarterly Reports. They will no longer be documented in the Unit 1 Recovery Quarterly Integrated Reports such as this one. Inspection Report 05000259,260,296/2006003, issued July 26, 2006, is the most recent Unit 2 and 3 Integrated Quarterly Report. Although that report did not contain any site inspections in these cornerstones, they will continue to be documented in ROP integrated quarterly reports such as that one.

This inspection examined activities conducted under your Unit 1 license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license and also with fulfillment of Unit 1 Regulatory Framework Commitments. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. A significant portion of your engineering activities, Unit 1 Recovery Special Program implementation, and modification activities were reviewed during this inspection period and found to be effective with no significant problems identified. However, based on the results of this inspection, a Severity Level IV violation of NRC requirements was identified resulting from failure to install instrument tubing supports in accordance with design drawing requirements. However, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy.

If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Browns Ferry Nuclear Plant.

Overall, we primarily found only minor discrepancies, indicating that your oversight of recovery activities was generally effective. However, we will continue to monitor implementation of your corrective actions to address previously identified weaknesses in your System Return to Service process.

Based on current and previous inspections of Unit 1 Recovery activities associated with five of your Special Programs, the staff has concluded that your implementation of these Special Programs has been adequate and when fully implemented should satisfy NRC regulatory requirements and commitments in your regulatory framework letter dated December 13, 2002. These Special Programs include the areas of Fuses, Thermal Overloads, Cable Splices, Moderate Energy Line Breaks, and the Q-List. We do not anticipate additional inspections for these areas.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Malcolm T. Widmann, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket No. 50-259
License No. DPR-33

Enclosure: Inspection Report 05000259/2006007
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Distribution w/encl: (See page 4)

Report to Karl W. Singer from Malcolm T. Widmann dated August 10, 2006

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INTEGRATED INSPECTION REPORT 05000259/2006007

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-259
License No:	DPR-33
Report No:	05000259/2006007
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Unit 1
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	April 16, 2006 - July 15, 2006
Inspectors:	W. Bearden, Senior Resident Inspector, Unit 1 E. Christnot, Resident Inspector C. Stancil, Resident Inspector G. Dipaolo, Senior Resident Inspector, Brunswick (Section E8.3) S. Walker, Resident Inspector, McGuire (Sections E8.7, E8.8) G. MacDonald, Senior Reactor Inspector (Section E8.4, E8.5, E8.6) L. Mellen, Senior Reactor Inspector (Sections E1.14, E8.1, E8.2, E8.12, E8.13, E8.14, E8.15, E8.16, E8.17) J. Lenahan, Senior Reactor Inspector (Sections E1.8, E1.9) N. Merriweather, Senior Reactor Inspector (Sections E1.11, E1.12, E1.13) M. Maymi, Reactor Inspector (Sections E1.15, E8.9) J. Rivera-Ortiz, Reactor Inspector (Section E1.17) R. Rodriguez, Reactor Inspector (Sections E1.11, E1.12, E1.13)

Enclosure

A. Rogers, Reactor Inspector (Section E1.10)
N. Staples, Reactor Inspector (Section E1.6)
R. Chou, Reactor Inspector (Section E1.10)
J. Fuller, Reactor Inspector (Section E1.16, E8.10, E8.11)
B. Miller, Reactor Inspector (Section E8.11)

Approved by: Malcolm T. Widmann, Chief
Reactor Project Branch 6
Division of Reactor Projects

EXECUTIVE SUMMARY

Browns Ferry Nuclear Plant, Unit 1
NRC Inspection Report 05000259/2006007

This integrated inspection included aspects of licensee engineering and modification activities associated with the Unit 1 recovery project. This report covered a three month period of resident inspector inspection. In addition, NRC staff inspectors from the regional office conducted inspections of Unit 1 Recovery Special Programs in the areas of fuses; thermal overloads; cable splices; electrical cable installation/separation; large bore pipe and supports; Q-List; moderate energy line breaks; small bore piping and instrument tubing; instrument sensing lines; and open inspection items. The inspection program for the Unit 1 Restart Program is described in NRC Inspection Manual Chapter 2509. Information regarding the Browns Ferry Unit 1 Recovery and NRC Inspections can be found at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/bf1-recovery.html. Per the Partial Cornerstone Transition letter from the NRC to TVA dated December 29, 2004, four Reactor Oversight Process (ROP) Cornerstones (Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection) are monitored under the ROP baseline inspection program as of January 2005. Consequently, inspections for these cornerstones are integrated with Unit 2 and 3 ROP baseline inspections and are no longer documented in the Unit 1 recovery quarterly integrated reports such as this one, but in the Unit 2 and 3 Integrated Quarterly Reports.

Inspection Results - Engineering

- The inspector's review of four planned modification design change packages concluded that the design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements. The designs adequately addressed the changes needed to restore Unit 1 to current requirements. (Section E1.1)

- Modification installation activities associated with four permanent plant design changes were observed and found to be performed in accordance with the documented requirements. (Section E1.1)

- Activities associated with removal of seven temporary alterations which affected Residual Heat Removal Service Water, Raw Cooling Water, Turbine Generator Control , Reactor Recirculation, Control Rod Drive, and Reactor Protection systems did not cause any significant impacts on the operability of equipment required to support operations of Units 2 and 3. (Section E1.2)

- Activities associated with the system return to service process were being adequately implemented. System turnover walkdowns were performed in accordance with procedural guidance. The initial system turnover plant review and acceptance boards appeared effective. However, inspectors will continue to review written guidance and observe future boards for long term effectiveness. Additionally, observation of future system turnover activities will be required to determine adequacy of corrective actions associated with previously identified weaknesses in the licensee's process. (Section E1.3)

- Activities associated with the area turnover process did not cause any significant impact to the operability of equipment required to support operations of Units 2 and 3. However, the inspectors determined that, as in the licensee's system turnover process, the plant will need to provide focused followup on outstanding open punchlist items. The inspectors identified plant areas that contain select safety related systems for future NRC inspection efforts. Further independent NRC inspection will be needed to determine consistent program implementation and resolution of select punchlist items, specifically fire seal barriers. (Section E1.4)

- Implementation of restart testing activities continued to be acceptable. Minor deficiencies were identified during performance of testing which did not effect the results of the testing. Licensee processes were effective at identifying problems before components were placed in service. (Section E1.5)

- Based on current and previous reviews, the inspectors determined that implementation of four sub-programs for the Cable Installation Special Program was proceeding in accordance with licensee commitments and regulatory requirements. These sub-programs included sidewall pressure, cable pullbys, cable jamming, and vertical cable supports. Completed actions to address these issues for Unit 1 are consistent with those previously committed to and performed for Units 2 and 3. The inspectors concluded that no issues related to these sub-programs that would negatively impact the restart of Unit 1 were identified as the result of the above reviews. No further inspections are anticipated for these four sub-programs. However, implementation activities associated with the remaining cable installation sub-program, bend radius of medium voltage cables, along with the Cable Separation Special Program, will need further inspections by the NRC to verify corrective actions are in accordance with licensee commitments. (Section E1.6)

- Activities associated with master equipment list data updates for the Residual Heat Removal Service Water and Emergency Equipment Cooling Water systems were adequately input and the database accurately represented components installed in the field. Based on the above review, the inspectors determined that the licensee's program for development of the Unit 1 Q-List was being performed in accordance with the documented requirements. Completed or planned actions were consistent with those previously performed for Units 2 and 3. However, only a small portion of the expected master equipment list updates have been completed; those updates will continue to be reviewed by the inspectors as part of future reviews of the system turnover process. No further inspections of this Special Program are anticipated. No violations or deviations were identified. (Section E1.7)

- Small Bore Piping support activities were performed in accordance with documented requirements with the exception of an example of a failure to construct a small bore pipe (instrument tubing) support in accordance with drawing requirements. This finding is one of two examples of a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V identified for failure to construct instrument tubing supports in accordance with design drawings requirements. The inspectors determined that the licensee's program for correction of deficiencies identified in support of small bore piping, including instrument tubing, complies with the design criteria, commitments to NRC, and NRC requirements. However, additional samples of small bore piping in the reactor building (outside the drywell) will need to be inspected prior to closure of this Special Program. (Section E1.8)

- The Instrument Sensing Line slope corrective activities were performed in accordance with documented requirements. A second example of a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V was identified for failure to construct instrument tubing supports in accordance with design drawing requirements. Additional samples of instrument sensing lines will need to be inspected to verify the instrument line slope deficiencies were corrected and supports were installed in accordance with design requirements prior to closure of this Special Program. (Section E1.9)

- Based on independent walkdowns of pipe supports, a pipe support calculation, as-built support drawings, and problem resolution, the inspectors determined that licensee performance was adequate in the Large Bore Pipe Support Special Program. However, additional samples will need to be inspected prior to closure of this Special Program. (Section E1.10)

- The inspectors concluded that the licensee's corrective action program to resolve the problems with misapplication of current-limiting fuses is acceptable to support Unit 1 restart. The program is equivalent in scope to those previously applied to the restart of the other units at Browns Ferry. The inspectors examined a select sample of the replacement fuses and verified that the program was being adequately implemented. Therefore, no further inspections of this Special Program are anticipated. (Section E1.11)

- Ongoing activities associated with the electrical terminations using Raychem splices were conducted in accordance with existing requirements. Additionally, the inspectors concluded that the special program for electrical cable splices and terminations in Equipment Qualification (EQ) applications was adequate to support Unit 1 restart. The program will replace most of the EQ splices on Unit 1. The inspectors confirmed, by examination of a select sample of completed splices and by witnessing two in-process splices, that the program was being adequately implemented. Therefore, no further inspections of this Special Program are anticipated. (Section E1.12)

- The inspectors concluded that the licensee's program to resolve problems with sizing of thermal overloads is acceptable to support Unit 1 restart. The licensee has replaced or strapped out the Unit 1 safety-related thermal overloads in accordance with the plant design. Therefore, no further inspections of this Special Program are anticipated. (Section E1.13)

- Based on current and previous inspections of Unit 1 Recovery activities associated with the Moderate Energy Line Break Special Program, the staff has concluded that the implementation of this Special Program has been adequate and should satisfy NRC regulatory requirements and commitments in the regulatory framework letter dated December 13, 2002. We do not anticipate additional inspections for this area. (Section E1.14)

- The licensee's Heat Sink Program was being adequately maintained. All changes to procedures and to the program were being performed in accordance with licensee commitments and NRC requirements. Based on focused reviews for Unit 1, inspectors did not identify any impediments to the future transition of Unit 1 Heat Sink inspections under the Initiating Events and Mitigating Systems' Cornerstones to the Reactor Oversight Process. (Section E1.15)

- The inspectors concluded that the licensee's Inservice Inspection (ISI) program was being adequately maintained. Changes to the program since the last inspection were consistent with licensee commitments and NRC requirements. Based on this review and past reviews of the Unit 1 ISI program, inspectors did not identify any impediments to the planned transition of Unit 1 ISI inspections under the Initiating Events, Barrier Integrity, and Mitigating Systems Cornerstones to the ROP. (Section E1.16)

- ASME Section XI Repair/Replacement activities were performed in accordance with the documented requirements. (Section E1.17)

<u>Inspection Results - Maintenance</u>

- The Maintenance organization continued to provide appropriate and comprehensive repairs to Unit 1 components which did not require design changes to support Unit 1 Restart. Work order packages included sufficient technical guidance to allow personnel to adequately perform the associated work activity. Maintenance personnel and foremen were knowledgeable of applicable requirements and appropriately documented work actually performed, as required by plant procedures. (Section M1.1)

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REPORT DETAILS

Summary of Plant Status

Unit 1 has been shut down since March 19, 1985, and has remained in a long-term lay-up condition with the reactor defueled. The licensee initiated Unit 1 recovery activities to return the unit to operational condition following the TVA Board of Directors decision on May 16, 2002. During the current inspection period, re-installation of plant equipment and structures continued. Recovery activities include ongoing replacement of small bore piping and instrument tubing in the drywell and reactor building; re-installation of balance-of-plant piping and turbine auxiliary components; installation of small and large bore pipe supports; and installation of new electrical cables, conduits, and conduit supports. The amount of restart testing and system return to service activities increased during this reporting period as the Unit 1 recovery effort continued to transition away from bulk construction work.

II. Engineering

E1 Conduct of Engineering

E1.1 Permanent Plant Modifications (71111.17, 37550, 37551)

a. Inspection Scope

In order to have some oversight of licensee recovery activities not directly limited to specific Unit Restart List Items, the inspectors reviewed planned Design Change Notice (DCN) packages associated with modifications to the Control Rod Drive (CRD) System, High Pressure Coolant Injection (HPCI) System, 120 VAC Electrical Distribution, and Appendix R Lighting. The inspectors reviewed criteria in licensee procedures Standard Program and Process (SPP)-9.3, Plant Modifications and Engineering Change Control; SPP-7.1, Work Control Process; SPP-8.3, Post-Modification Testing; and SPP-8.1, Conduct of Testing, to verify that risk-significant plant modifications were developed, reviewed, and approved per the licensee's procedure requirements.

The inspectors reviewed and observed ongoing modification activities to various electrical components in the Residual Heat Removal (RHR) System, HPCI System, 48 VDC distribution, and various Control Room Design Review (CRDR) modification activities. The inspectors evaluated the adequacy of the modifications and observed field work to verify that the design basis, licensing basis, and Technical Specification (TS) requirements for the systems had not been degraded as a result of the modifications.

b. Observations and Findings

b.1 DCN Package Review

The inspectors reviewed the following DCNs associated with planned modifications on Unit 1 to verify that the packages contained adequate design information and supporting analyses to allow modifications personnel to properly implement the desired change, update plant documentation, and resolve the identified condition. In addition, the inspectors verified that the planned modifications would not adversely affect the design basis of the system or interfacing systems. Also, the inspectors verified that the planned modifications would not place either of the operating units in an unsafe condition.

DCN 51198

The inspectors reviewed permanent plant modification DCN 51198, High Pressure Coolant Injection (HPCI) Mechanical - Reactor Building, System 73. The intent of this DCN was to implement the mechanical modifications recommended for the HPCI system in the reactor building. The DCN consisted of two stages and included various work activities involving HPCI mechanical components. Planned changes included installation of live load packing on valves 1-FCV-73-26, 1-FCV-73-27, 1-FCV-73-34, and 1-FCV-73-356; replacement of various drain valves; installation of new motor operators on valves 1-FCV-73-16, 1-FCV-73-34, 1-FCV-73-36, and 1-FCV-73-40; removal and replacement of existing valve 1-FCV-73-03 and live load packing; and removal and installation of various new drain and test valves. Additionally, this DCN replaced existing EGR type hydraulic actuator 1-SM-73-19 for valve 1-FCV-73-19; replaced existing obsolete valve 1-RFV-73-506 and associated piping; replaced testable check valve 1-FCV-CHK-73-45 along with the associated pneumatic operator; replaced solenoid valves 1-FSV-73-17A and 1-FSV-73-17B; replaced existing HPCI Booster Pump rotor with a new rotor assembly; and performed work as required to seismically qualify the HPCI turbine pump skid and auxiliary components per GE Specification FDI 171-10500. The DCN also required testing of various system components including hydrostatic testing of components, dynamic testing of components in the steam and water handling portions of the system, and MOVATS testing on identified valves.

DCN 51206

The inspectors reviewed permanent plant modification DCN 51206, Control Rod Drive (CRD) Electrical and Mechanical - Reactor Building, System 85. The intent of this DCN was to implement the electrical and mechanical modifications recommended for the CRD system in the reactor building. The DCN consisted of four stages and included various CRD valves and electrical components. Planned changes included replacement of valves 1-RFV-85-505A, 1-ISV-85-586, 1-FCV-85-54, 1-FCV-85-11A, 1-FCV-85-55, and 1-FCV-85-11B; replacing packing on valves 1-FCV-85-56, 0-SHV-85-500, and 1-BYV-85-519A; installing new relief valves 1-85-RFV-604 and 1-RFV-85-609; and modifying and installing doors and cages on the existing east and west Scram Discharge Instrument Volume (SDIV) cages. This DCN also determinated and removed various cables and affected conduit; removed lighting circuits and conduit from panel 1-25-48A; removed and discarded panels 1-PNLA-925-48A and 1-PNLA-925-48B; installed level switches 1-LS-85-45C, 1-LS-85-45D, 1-LS-85-45E and 1-LS-85-45F on the SDIV; disabled low scram pilot air header pressure switches 1-PS-85-35A1, 1-PS-85-35A2, 1-PS-85-35B1, and 1-PS-85-35B2; installed new SDIV valves 1-RTV-85-227A, 1-RTV-85-278A, 1-RTV-85-286A, and 1-RTV-85-288A; rebuilt SDIV test Manometer;

and installed various supports on CRD piping in the reactor building. The DCN also required that setpoint and scaling documents and activities be performed on various instrumentation during the performance of Post Modification Testing (PMT). The inspectors noted that Work Order (WO) 03-006734-22 was issued to control any needed system turnover punchlist items for setpoint and scaling issues identified during PMT activities.

DCN 51214

The inspectors reviewed permanent plant modification DCN 51214, 120V AC Distribution Electrical - Reactor Building, System 57-2. The intent of this DCN was to implement the electrical modifications recommended for the 120V AC system in the reactor building and in support of new inverters 1-INVT-255-01 and 1-INVT-256-02. The DCN consisted of two stages. Planned electrical modifications included determination, lifting, and pulling out old cables at various locations on the 621 ft elevation of the reactor building; reworking, removing, or abandonment in place of conduits at various locations in the reactor building; installation of new conduits and supports; and pulling and termination of new cables in 1-PNLA-09-81 and 1-PNLA-09-87. The DCN also required core drilling be performed from the reactor building into electrical board rooms on elevations 593' and 621'. The core drilling activities were to be controlled by WO 03-004725-16 and WO 03-004725-17.

DCN 51229

The inspectors reviewed permanent plant modification DCN 51229, Appendix R and 240V Emergency Lighting - Reactor Building, System 999 and System 247. The intent of this DCN was to implement the electrical modifications recommended for the emergency lighting system in the reactor building. The DCN consisted of 13 stages and included fabrication and installation of battery pack supports for Appendix R emergency lights, installation of battery packs with attached lights, and incorporation or verification of non-scope items for the Master Equipment List (MEL). The DCN also required that selected existing non-Appendix R battery pack emergency lights in the reactor building be replaced. The non-Appendix R battery pack activities were to be controlled by WOs 03-015067-03, 03-015067-05, 03-015067-09, and 03-015067-10.

b.2 Implementation of Permanent Plant Modifications

The inspectors reviewed selected portions of the following ongoing modifications on Unit 1 to verify adequacy of the modifications and observed field work to verify that the design basis, licensing basis, and TS requirements for the systems had not been degraded as a result of the modifications.

DCN 51106

The inspectors reviewed and observed portions of permanent plant modification activities associated with DCN 51106, Control Room Design Review (CRDR) - Control Bay, Stage 1. This stage involved the Control Panel 1-25-32 (Remote Shutdown Panel) located in Shutdown Board Room A. The modification activities involved instrumentation changes for System 64, Containment; System 71, Reactor Core Isolation Cooling (RCIC); System 74, Residual Heat Removal (RHR); and System 256, Emergency Core Cooling System (ECCS) Inverters. Work was controlled by WO series 02-011701-03 thru 16. Work activities observed included selected portions of the installation of new containment control instrumentation including 1-IS-64-67A, 1-PS-64-67B, 1-PX-64-50, 1-TM-64-52AA, and 1-TS-64-052A; relocate and replace of hand switch handles, replace indicating light lenses, replace annunciator window 1-ZA-71-49B, change indicator scale for 1-SI-71-42B, and replace 1-FIC-71-36B with a Yokogawa digital controller for RCIC. Observations also included replacement of meter and scale for 1-FI-74-79, and installation of square root transmitter 1-FM-74-79 for RHR; and installation of new transfer switch 1-XS-256-1 for the ECCS inverters. The ongoing modification activities affected the operating units and required entry into a 30-day Limiting Condition for Operations (LCO). The inspectors verified that all work activities were completed within the 30 day limit.

DCN 61728

The inspectors reviewed and observed portions of permanent plant modification activities associated with DCN 61728, 48 VDC Distribution - Control Bay, System 57-6, Stage 1, and Stage 2. These stages involved the 48V DC distribution for the following equipment: Neutron Monitoring Battery Charger A1-1, Neutron Monitoring Battery Charger A1-2, Neutron Monitoring Battery Charger B1-1, and Neutron Monitoring Battery Charger B1-2. The activities were controlled by WOs 04-724031-04 and 04-724031-09. Work activities observed included selected portions of the removal of the existing battery chargers and installation of new battery chargers.

DCN 51198

The inspectors reviewed and observed portions of the permanent plant modification activities associated with DCN 51198, HPCI Mechanical - Reactor Building, System 73, Stage 1. This stage involved the HPCI valve 1-FCV-73-36, HPCI Test Return to Unit 1 Condensate Storage Tank (CST). The modification activities were controlled by WO 03-000997-33. Work activities observed included selected portions of the removal and relocation of the valve (cutting out of the valve), installation of a new piping spool piece, welding in the spool piece, and welding the valve back in at the new location. The valve was relocated approximately ten feet from the original location and still within the Unit 1 HPCI room. The test return line to the CST involved a penetration through secondary containment which required plugging during a portion of the ongoing work activities. The inspectors verified that the plug used in conjunction with the allowable secondary containment breach size did not violate secondary containment integrity.

DCN 51222

The inspectors reviewed and observed portions of permanent plant modification activities associated with DCN 51222, RHR Electrical - Reactor Building, System 74, Stage 2. This stage involved electrical cables associated with RHR Pump 1B. The activities were controlled by WO 03-000997-33. Work activities observed included selected portions of the installation and termination of the new cables, tagging of the new cables with identification labels, and replacement of contact blocks for local hand switch 1-HS-74-28B. The work activities were performed inside the 4KV Shutdown Board A.

c. Conclusions

The inspectors' review of modification design packages associated with four DCNs concluded that the design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements. The DCNs adequately addressed the changes needed to restore Unit 1 to current requirements.

Modification activities associated with four ongoing permanent plant modifications were performed in accordance with the documented requirements.

E1.2 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed licensee procedure SPP-9.5, Temporary Alterations. The inspectors also reviewed and observed ongoing activities associated with System 23, Residual Heat Removal Service Water (RHRSW); System 24, Raw Cooling Water (RCW); System 47, Turbine Generator Control (EHC); System 68, Reactor Water Recirculation (RWR); System 85, Control Rod Drive (CRD); and System 99, Reactor Protection (RPS). The inspectors verified that 10 CFR 50.59 screening and technical evaluations against the system design bases documentation, including the Final Safety Analysis Report (FSAR) and Technical Specifications and reviewed selected completed work activities of the system to verify that installation and/or removal was consistent with the modification documents and the Temporary Alteration Control Form (TACF). In addition, special emphasis was placed on the potential impact of these temporary modifications on operability of equipment required to support operations of Units 2 and 3.

b. Observations and Findings

The inspectors reviewed and observed selected portions of ongoing activities associated with temporary alterations for ongoing removal activities associated with temporary alterations involving temporary RHRSW inlet bay supply line sluice gate closure limitations; main generator sample valve; CRD pressure gauges, flanges, and unions; RCW piping plug; and RPS jumpers and fuses. The inspectors verified that the ongoing temporary modification activities were consistent with the applicable documentation, configuration control of the temporary modification was adequate, and post-installation testing confirmed actual impact of the modification on permanent systems and interfacing systems. In addition, the inspectors verified that the activities did not cause an adverse impact on operability of structures, systems, and components (SSCs) required to support operations of Unit 2 and 3. The temporary alterations reviewed and observed were as follows:

<u>TACF 1-84-050-47</u>

This TACF had been initiated to install a grab sample valve in the Unit 1 cooling water system for the main generator output breaker 1-PCB-35-214. The valve was installed between pressure indicator 1-PI-47-201 and shut off valve 1-SHV-47-201 in System 47, EHC. The inspectors reviewed DCN 51126, Water Quality and Sampling (WQS) - Turbine Building, System 43, and verified that this DCN made this change permanent. WO 06-711346-00 was used to remove the TACF. The DCN also changed the system designation from System 47, EHC, to System 43, WQS; designated the valve as 1-SMV-43-852; and revised drawing 1-HAMT301456-1.

<u>TACF 1-85-002-85, Revision Rev 1</u>

This TACF was originally initiated to document pressure gauges in the CRD system that were installed on the strainers in the suction side of the 1A and 1B CRD pumps. The gauges were installed to verify the condition of the strainers. The inspectors reviewed DCN 51240, CRD - Reactor Building, Stage 4, and verified that this DCN made this change to the 1A pump permanent. WO 04-718067-04 was used to remove the TACF. The part of the TACF affecting the 1B pump had previously been made permanent by DCN H1359 in 1990.

<u>TACF 1-85-007-68</u>

This TACF was initiated to cut out the 3/4 inch vent line in RCW system valve 1-FCV-68-03, RCW Pump 1A Discharge, install a tapered stainless steel plug, and seal weld the plug. The valve was removed and replaced as part of the Unit 1 RCW system replacement. The inspectors reviewed WO 02-010314-02 and verified that this WO was used to document the removal the TACF.

<u>TACF 1-85-030-68</u>

This TACF was issued to document the existence of flanges and unions in the injector water relief piping of the CRD system that are not shown on the mechanical drawings. It was documented that the flanges and unions were rated to the same limits as the piping. They did not affect the rating or the integrity of the piping. The inspectors reviewed Mechanical Drawing 47W465-4 and noted that it was revised to make the change permanent. WO 02-016506-10 was used to remove the TACF.

TACF 1-85-032-99

This TACF was issued in October 1985 to install jumpers in the RPS system Channels A and B to prevent scrams during the HFA relay coil change out. The original jumpers used alligator clips. During the Unit 1 recovery, Problem Evaluation Report (PER) 46920 was generated to document the use of alligator clips. As a result, WO 04-717791-00 was initiated to remove the clips and replace them with wired jumpers. The inspectors reviewed DCN 51080, RPS - Control Bay, System 99, and verified that this DCN replaced the HFA relays in the RPS. The inspectors also reviewed WOs 03-002046-00 and 03-002048-00 and verified these WOs were used to remove the TACF.

TACF 1-85-016-99

This TACF was issued in July 1986 to remove fuses in the RPS system Channels A and B to inhibit the backup scram function during the HFA relay coil change out. The fuses were 5A-F21A and 5A-F22A in Control Panel 1-9-15 and fuses 5A-F21B and 5A-F22B in Control Panel 1-9-17, located in the Unit 1 Auxiliary Instrument Room. The inspectors reviewed TACF 1-04-010-99 which was subsequently issued in September 2004 and noted that this newer TACF also required that these fuses be removed. Consequently, TACF 1-85-016-99 was administratively closed and TACF 1-04-010-99 remained in effect.

TACF 0-04-004-023

This TACF was previously issued for RHRSW inlet bay supply line sluice gate closure to support work activities associated with Unit 1 Condenser Circulating Water (CCW), System 27, by providing an alternative set of limitations for closure of sluice gates for the supply lines from inlet bays to the RHRSW pump suction pit during two unit operations. The installation of TACF 0-04-004-023 was documented in NRC Inspection Report 259/2004-007. The TACF specifically revised Note 7B on Drawing 1-47E858-1 to read: During two-unit operation, two gates, but not two on the same supply line tee may be closed at any time, except two gates on the same supply line tee may be closed as stated in Note 8. The new Note 8 read: To ensure capability to safely shutdown and proper RHRSW pump operation, following a down stream dam break with two units operating and two sluice gates closed on the same RHRSW pump pit supply line, all of the following limitations shall apply:

- Each unit shall be operated at less than or equal to 3458 MWT.
- This alignment shall only be utilized while river temperature is less than or equal to 91 degrees F.
- RHRSW flow is restricted to less then or equal to 4100 gpm per pump, 2 pumps per unit.

- Emergency Equipment Cooling Water (EECW) is reduced to two pumps at less than or equal to 4500 gpm per pump prior to starting RHRSW pumps.

The inspectors reviewed selected removal activities for this TACF. Additionally, selected completed system work activities were verified to be installed and removed consistent with the modification documents and TACF. The inspectors also verified adequate configuration control through updated plant documents, drawings, and procedures, and confirmed satisfactory post-installation and removal test results.

c. Conclusions

The inspectors determined that activities associated with removal of seven temporary alterations which affected RHRSW, RCW, EHC, RWR, CRD, and RPS systems did not cause any significant impacts on the operability of equipment required to support operations of Units 2 and 3. No violations or deviations were identified.

E1.3 System Return to Service Activities (37550, 37551)

a. Inspection Scope

The inspectors continued to review and observe portions of the licensee's ongoing System Return to Service (SRTS) activities. The SRTS activities were performed in accordance with Technical Instruction 1-TI-437, System Return to Service Turnover Process for Unit 1 Restart. The level of SRTS activities continued to increase during this reporting period as the Unit 1 recovery effort continued to transition away from bulk construction work. However, only a limited number of important risk significant systems have completed SRTS activities.

Additionally, the inspectors observed several System Pre-Operability Checklist (SPOC) plant system walkdowns and plant acceptance boards. SPOC I walkdowns were observed on System 70, Reactor Building Closed Cooling Water (RBCCW), and System 57-2, 120 VAC Instrument and Control Power. A SPOC II walkdown was observed on System 80, Primary Containment Temperature Monitoring. Inspectors observed SPOC plant acceptance boards on RBCCW; Primary Containment Temperature Monitoring; System 57-4, 480 VAC Distribution; and System 85, Control Rod Drive.

b. Observations and Findings

The SRTS process consisted of three parts: System Plant Acceptance Evaluation (SPAE), which consists of verification of design changes, engineering programs analysis, drawings, calculations, corrective action items, and licensing issues; SPOC I, which consists of the completion of items required for system testing; and SPOC II, which consists of the completion of system testing and the completion of items that affect operational readiness. All required system SPAE packages had previously been issued by the licensee prior to the start of this reporting period.

Specific SRTS activities observed by the inspectors included periodic meetings to discuss the SRTS status, which included the status of the SPOC I checklists, status of the SPOC II process, and status of outstanding work items and identified deficiencies. Documents and activities reviewed included System SPOC exceptions, deferrals, and special operating conditions; system testing requirements; temporary alterations; completed work orders (WOs); engineering calculations; SRTS open items punchlist (OIP); and various PERs associated with the SRTS process. The inspectors also held discussions with engineering and operations personnel responsible for SRTS activities and performed walkdowns of selected portions of affected systems.

SPOC walkdowns observed by the inspectors were adequately scheduled and conducted in accordance with Technical Instruction 1-TI-437, Appendix D, SPOC Walkdown Instruction, and Attachment 6, SPOC Walkdown. Pre-walkdown briefs were held by the restart system engineer for all participants where walkdown boundaries and guidelines were discussed. The minimum required representatives were present and middle level management accompanied several of the walkdowns. Operational deficiencies and material discrepancies identified by inspectors were independently identified by the licensee and appropriately documented and dispositioned. In particular, one potential bend radius issue identified by inspectors during the System 80 walkdown was adequately resolved by the licensee.

The licensee continued to address SRTS weaknesses previously identified by the inspectors and by the licensee during a licensee self assessment. NRC inspection findings related to those weaknesses were previously discussed in Inspection Report 50-259/2006-06. Improvements to the licensee's SRTS process were initiated to address those weaknesses. These improvements included increased management expectations regarding ownership by personnel from the operating organization, greater level of involvement by management (both from Unit 1 and the operating units), and creation of separate plant review and acceptance boards tasked with providing independent oversight of SPOC I and SPOC II turnover activities for each system which undergoes SRTS activities. The inspectors concluded that the initial SPOC boards appear to be effective for plant determination of SPOC I and II acceptance and were appropriately represented by plant and restart organizations. However, written procedural guidance was still being developed. Additionally, the inspectors intend to conduct future inspection of licensee corrective actions taken as a result of inspection findings referenced in Inspection Report 50-259/2006-06.

c. Conclusions

Inspectors determined that activities associated with the SRTS turnover process were being adequately implemented. SPOC walkdowns were performed in accordance with procedural guidance. The initial SPOC plant review and acceptance boards appeared effective. However, inspectors will continue to review written guidance and observe future boards for long term effectiveness. Additionally, observation of future SRTS activities will be required to determine adequacy of corrective actions associated with previously identified weaknesses in the licensee's SRTS process.

E1.4 Area Turnover Activities (37550, 37551)

a. Inspection Scope

The inspectors reviewed TVA Business Practice BP-338 which describes the licensee's program for area turnovers performed during the Unit 1 Restart Recovery Project. Additional document reviews included area turnover packages comprised of the remaining punch listed action items, unfinished scheduled work activities, and previous walkdown items. The inspectors interviewed the program developer and procedure writer, Unit 1 Area Turnover Coordinator, and designated area coordinators, including the operations focal point of contact. The area turnover schedule and status was reviewed for rationale and project integration. Several plant area walkdowns, during both preliminary and final acceptance by the licensee, were conducted by the inspectors. Independent inspector walkdowns were performed to focus on area deficiencies. Additionally, observations of plant and restart management were performed to determine licensee area turnover philosophy and methodology.

b. Observations and Findings

The inspectors determined that area turnover guidance and turnover packages submitted to walkdown participants were adequate. Though very early in the process, the inspectors determined that areas are being turned over with Unit 1 restart work still outstanding in most areas. As of the end of this reporting period only two areas had been accepted by plant operating organization. Those two areas were Reactor Building 639 North and 639 South. Outstanding work and other deficiencies are being identified by restart area coordinators and plant management during walkdowns and subsequently punchlisted in a similar process as used for SPOC II system turnover. The inspectors determined that the most significant outstanding work items to date were the lack of completion of fire barrier seals in the turned over areas. These activities will be carried as open on the punch lists until the fire seal DCNs are completed. This effort is currently targeted for fuel load. In addition, because this system (System 100, Penetrations and Sleeves) is not planned to separately undergo the SPOC process, significant emphasis will be placed on verification of new, modified, repaired/replaced, or deleted fire zone penetrations. Plant Fire Operations intends to accept all Unit 1 fire zone fire barrier seals as fully ready upon the restart organization completion (RTO, return to operations) of associated DCNs and work orders. Plant Fire Operations intends to then implement their required surveillances for operability which are barrier-based (floor, ceiling, wall), not individual penetration-based. The inspectors reviewed select penetration fire seal installation work orders and associated data sheets and determined that the DCN, work order process, and maintenance procedures provided adequate craft and quality control verification of fire barrier seals. Upon DCN completion, the restart system engineer is planning a 100% fire barrier seal walkdown with a multi-disciplined group from plant and restart organizations.

Enclosure

The inspectors questioned the adequacy of housekeeping controls once area responsibilities are assumed by the plant organization. Of the two formal area turnovers completed, neither had been added to the plant housekeeping zone responsibility list. Plant maintenance management informed the inspectors that a process for controling housekeeping was under development.

The inspectors noted that the licensee is leaving the orange unit separation tape (defining boundaries between the operating units and Unit 1) in place due to licensee identified labor intensive removal requirements. The inspectors determined separation tape was a cosmetic issue and not a functional impact to areas, systems, or components.

c. Conclusions

The inspectors determined that activities associated with the Unit 1 restart area turnover process did not cause any significant impact to the operability of equipment required to support operations of Units 2 and 3. However, the inspectors determined that, as in the previous NRC review of the SPOC process, the plant will need to provide focused followup on outstanding open punch list items. The inspectors have identified plant areas that contain select safety-related systems for future independent NRC inspection and intend to continue inspection efforts to determine consistent program implementation and resolution of select punchlist items, especially fire seal barriers. No violations or deviations were identified.

E1.5 System Restart Testing Program Activities (37551)

a. Inspection Scope

The inspectors reviewed and observed on-going Restart Test Program (RTP) activities associated with post modification testing for five risk significant systems to ensure activities were in compliance with design basis requirements. Additionally, the inspectors reviewed RTP activities associated with testing for fire protection systems to ensure activities were in compliance with design basis requirements.

b. Observations and Findings

b.1 Post Modification Testing Activities

Post modification testing activities reviewed and observed consisted of post modification testing performed on System 23, Residual Heat Removal Service Water Cooling Water (RHRSW); System 64C, Secondary Containment; System 65, Standby Gas Treatment (SBGT); System 57-4, 480V Electrical System; and the Common Accident Signal (CAS) Logic.

Test procedures consisted of Post Modification Test Instructions (PMTIs) and were issued to test portions of applicable DCNs. The inspectors verified that pre-test briefings were held, assignments made, and communications were established prior to performance of testing. Specific post modification testing activities reviewed and observed included the following:

<u>1-PMTI-BF-51090-STG53, Rev. 0</u>

This testing satisfied the post modifications test requirements for Stage 53 of DCN 51090, 480V Electrical System - Control Bay, System 57-4. This DCN is part of the load shed program and was intended to ensure that the diesel generators (DG) will not overload during a design basis event with the addition of the Unit 1 electrical loads for restart. This stage consisted of addition of handswitch 0-HS-31-2101E, Load Shed Bypass, on Control Bay HVAC Panel 0-LPNL-925-165D and the re-wiring of transfer switch 0-XS-31-2101 for normal and remote operation to satisfy Appendix R requirements. The objective of this test was to demonstrate that the changes made to the 480V Load Shed Logic System by this DCN performed their intended function. Testing was performed by actuation of a simulated Division I actuation of the A2 Unit 1 480V load shed logic relay contacts and verification of trip of the Chilled Water Pump A. Then, with the simulated load shed A2 signal in place, the new bypass switch was placed in the BYPASS position and the Chilled Water Pump A was re-started. A similar test was performed with a simulated Division II actuation of the B2 Unit 1 480V load shed logic relay contacts. The inspectors observed portions of ongoing testing, reviewed the completed test package, verified acceptance criteria were satisfied, and verified testing successfully fulfilled the testing requirements for portions of work performed under Stage 53. There were no test exceptions.

<u>1-PMTI-BF-51102-STG02, Rev. 1 and Rev. 2</u>

This testing satisfied the post modifications test requirements for Stage 2 of DCN 51102, Control Room Panel 1-9-25. This DCN is part of the CRDR program for the Secondary Containment System (System 64C). The stage consisted of modifications to control switches located on Main Control Room (MCR) Panel 1-9-25. Modifications relocated handswitch 1-HS-64-10, Refuel Zone Exhaust Duct Inboard Isolation Damper; rewired handswitch 1-HS-64-11A, Reactor Zone Fans and Dampers; relocated handswitch 1-HS-64-14, Reactor Zone Air Supply Inboard Isolation Valve; relocated handswitch 1-HS-64-41, Reactor Zone Exhaust to Standby Gas Treatment System; relocated handswitch 1-HS-64-42; and relocated handswitch 1-HS-64-45, Refuel Zone Exhaust to Standby Gas Treatment System. This stage also relocated the associated handswitch indicating lights. Revision 1 of the test corrected the location of the breakers associated with the ventilation system. Revision 2 of the test corrected the old handswitch positions to the new positions. The objective of this test was to demonstrate that the changes made to the handswitches by this stage of the DCN did not affect the intended functions. Testing was performed by manually manipulating the various handswitches and verifying that the affected equipment functioned properly. The inspectors reviewed the completed test package and verified acceptance criteria for the test were satisfied. There were two test discrepancies (TDs) identified during the ongoing testing. The first involved handswitch 1-HS-64-11A, Reactor Zone Fans and Dampers, in that the positions Fast A and Fast B did not match the procedure; the second involved Damper 1-FCO-64-42 which had a leaking isolation valve and allowed to damper to drift open. The inspectors verified that these discrepancies were corrected prior to continuing with testing. The inspectors determined that the testing successfully fulfilled the testing requirements for portions of work performed under DCN 51102 Stage 2. There were no test exceptions.

<u>1-PMTI-BF- 51102-STG01, Rev. 1 and Rev. 2</u>

This testing satisfied the post modifications test requirements for Stage 1 of DCN 51102, Control Room Panel 1-9-25. This DCN is part of the CRDR program for the Secondary Containment System. The stage consisted of modifications to handswitches located on MCR Panel 1-9-25 and were the similar to those tested in Stage 2. Testing for Stage 1 tested Division I handswitches which controlled the outboard isolation dampers, where as Stage 2 tested Division II handswitches which controlled the inboard isolation dampers. The objective of this test was to demonstrate that the changes made to the Division I handswitches by this stage of the DCN did not affect the intended function. The test was performed by manually manipulating the various handswitches and verifying that the affected equipment functioned properly. The inspectors observed portions of the ongoing testing, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. There was one TD identified in that handswitch 1-HS-64-3A, Reactor Zone Fans and Dampers, was installed upside down. The discrepancy was corrected prior to continuing testing. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed under DCN 51102 Stage 1. There were no test exceptions.

<u>1-PMTI-BF- 51018-STG03, Rev. 1</u>

This testing satisfied the post modifications test requirements for Stage 3 of DCN 51018, Common Accident Signal Logic - Unit 2. This DCN is part of the Unit 1/2 Common Accident Signal (CAS) logic required for Unit 1 restart. Stage 3 consisted of modifications to install a new isolation junction box, termination block, and new cables affecting the Unit Priority Re-trip logic circuit associated with the B DG breaker 1822, DG output breaker to the 4KV Shutdown Board B. Testing was performed by initiation of a simulated CAS logic actuation at Panel 0-LPNL-925-0045B, 4KV Shutdown Board B Logic Panel, located in the 4KV shutdown board room B, and verifying that relay 0-RLY-211-TSCRNB, located in the same panel, energized. Additionally, when relay 2-RLY-074-10AK134A, DG B Unit Priority Retrip Breaker 1822 (Division I) at Panel 2-9-32, located in the Unit 2 Auxiliary Instrument Room, was manually actuated, relay 0-RLY-211-TSCRNB was verified as de-energized. The inspectors observed portions of the ongoing test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed under DCN 51018, Stage 3. There were no test exceptions.

1-PMTI-BF- 51102-STG03, Rev. 0;- STG04, Rev. 0; and- STG15, Rev. 0

This series of PMTIs satisfied the post modifications test requirements for Stages 3, 4, and 15 of DCN 51102, Control Room Panel 1-9-25. The DCN is part of the CRDR program for the System 64C Secondary Containment,. The stages consisted of modifications to handswitches located on MCR Panel 1-9-25 and were the similar to those tested in Stage 1 and Stage 2. The stages consisted of relocating handswitch 1-HS-64-06, Refuel Zone Air Supply Inboard Isolation Valve; handswitch 1-HS-64-43, Reactor Zone Air Supply Outboard Isolation Valve; and handswitch 1-HS-64-120, Reactor Zone Outboard Isolation Logic Test Switch. The objective of the tests was to demonstrate that the changes made to the various handswitches by the stages of the DCN did not affect the intended functions. The tests were performed by manually manipulating the various handswitches and verifying that the affected equipment functioned properly. The inspectors reviewed the completed test package and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for portions of work performed under DCN 51102 Stage 3, Stage 4, and Stage 15. There were no test exceptions.

1-PMTI-BF-51177- STG05, Rev. 0

This PMTI satisfied the post modifications test requirements for Stage 5 of DCN 51177, RHRSW Mechanical - Reactor Building, System 23. The DCN is part of the RHRSW upgrade program. Modifications consisted of changes to control and power cables to valve 1-FCV-23-46, RHR Heat Exchanger 1B RHRSW discharge, and to valve 1-FCV-23-57, standby coolant supply from the D header of the RHRSW system to Unit 1 RHR system Loop II and to Unit 2 RHR system Loop I. Among the objectives of this test were to demonstrate: proper opening and closing of valve 1-FCV-23-46 by manually operating handswitch 1-HS-23-46A on MCR Panel 1-9-3, by manually operating local control push buttons 1-HS-23-46B, and by manually operating handswitch 1-HS-23-46C on Reactor Motor Operated Valve (RMOV) Board 1B; the valve interlocks associated with RHRSW Pumps D1 and D2; the opening and closing of valve 1-FCV-23-57 by manually operating handswitch 1-HS-23-457A on MCR Panel 1-9-3, by manually operating local control push buttons 1-HS-23-57B, and by manually operating handswitch 1-HS-23-57C on RMOV Board 1B; the valve interlocks associated with RHRSW Pumps B1 and B2; and agreement between local and remote indicating lights and actual valve position. The test also demonstrated these valves could not be operated from the MCR when the respective transfer switches were placed in the EMERG position and could only be operated by the remote switches.

The inspectors observed portions of the ongoing testing, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. There were two TDs identified during the ongoing testing. The first TD involved valve 1-FCV-23-46 which did not function properly when the MCR handswitch 1-HS-23-46A was placed in the open position. WO 05-721298-00 was initiated to address this problem and a discrepancy was discovered in junction box 1-JBOX-303-11775. The discrepancy was corrected and the test was continued. The second TD involved valve 1-FCV-23-57 which did not function properly when the MCR handswitch 1-HS-23-57A was placed in the open position. WO 05-722836-00 was initiated to address this problem and a discrepancy was discovered in that the closed contact on the limit switch for valve 1-FCV-74-101, standby coolant supply from valve 1-FCV-23-57 of the RHRSW system to Unit 1 RHR system Loop II, had not been adjusted properly. The limit switch was adjusted and the test was continued. The inspectors determined that the testing successfully fulfilled the testing requirements for portions of work performed under DCN 51177, Stage 5. There were no test exceptions.

<u>1-PMTI-BF- 51102-STG08, Rev. 0, and STG 10, Rev. 0</u>

These PMTIs satisfied the post modifications test requirements for Stages 8 and 10 of DCN 51102, Control Room Panel 1-9-25. The DCN is part of the CRDR program for the SBGT System. Modifications consisted of changes to handswitches located on MCR Panel 1-9-25 and were the similar to those tested in Stages 1, 2, 3, 4, and 15. The stages consisted of relocating handswitch 0-HS-65-4A, SGT Train A Decay Heat Damper; handswitch 0-HS-65-25A, SGT Train B Inlet Damper, handswitch 0-HS-65-40A, SGT B Blower Auto/Manual Start; and handswitch 0-HS-65-48B, Keylock Test. The objectives of the tests were to demonstrate that the changes made to the various handswitches by the stages of the DCN did not affect the intended functions. The tests were performed by manually manipulating the various handswitches and verifying that the affected equipment functioned properly. The inspectors reviewed the completed test package and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for portions of work performed under DCN 51102 Stage 8 and Stage 10. There were no test exceptions.

<u>1-PMTI-BF- 51018-STG04, Rev. 1</u>

This PMTI satisfied the post modifications test requirements for Stage 4 of DCN 51018, CAS Logic - Unit 2. This DCN is part of the Unit 1/2 CAS logic required for Unit 1 restart. Stage 4 consisted of modifications to install a new isolation junction box, termination block, and new cables affecting the Unit Priority Re-trip logic circuit associated with the D DG breaker 1816, DG output breaker to the 4KV Shutdown Board D. Objectives of this test were to demonstrate that upon initiating a simulated CAS logic actuation at Panel 0-LPNL-925-0045D, 4KV Shutdown Board D Logic Panel, located in the 4KV shutdown board room D, relay 0-RLY-211-TSCRND, located in the same panel, energized; and when relay 2-RLY-074-10AK134B, DG D Unit Priority Retrip Breaker 1816 (Division II) at Panel 2-9-33, located in the Unit 2 Auxiliary Instrument Room, was manually actuated, relay 0-RLY-211-TSCRND de-energized. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed under DCN 51018, Stage 3. There were no test exceptions.

Enclosure

1-PMTI-BFN- 51177- STG06, Rev. 1

This PMTI satisfied the post modifications test requirements for Stage 6 of DCN 51177, RHRSW Mechanical - Reactor Building, System 23. The stage is part of the RHRSW upgrade program. The stage consisted of control cable and power cable modifications to valve 1-FCV-23-52, RHR Heat Exchanger 1D RHRSW discharge. The objective of this test was to demonstrate opening and closing of valve 1-FCV-23-52 by manually operating hand switch 1-HS-23-52A on MCR Panel 1-9-3, by manually operating local control push buttons 1-HS-23-52B, and by manually operating hand switch 1-HS-23-52C on RMOV Board 1B; the valve interlocks associated with RHRSW Pumps D1 and D2; and agreement between local and remote indicating lights and actual valve position. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed DCN 51177, Stage 6. There were no test exceptions.

b.2 Restart Testing Activities for Fire Protection Systems

A significant increase in the amount of testing of Unit 1 Fire Protection System equipment occurred during this reporting period. Testing observed by the inspectors was focused on the Fire Detection System (System 26). Restart testing consisted of PMTIs which were intended to satisfy the post modifications test requirements for DCN 51368, Unit 1 Fire Detection and Alarm, Stage 4. PMTIs observed and reviewed included:

1-PMTI-51368-STG04A, Rev. 0

This PMTI satisfied the post modifications test requirements for Stage 4A of DCN 51368. This stage involved modification activities to the local fire control panel 0-LPNL-25-544. The test consisted of panel initial inspection, panel checkout, initial panel power up, panel programming, and circuit testing. The test was performed per the EST-3 Vendor Installation and Service Manual with a vendor representative present. A 24-hour battery power test was also performed on the panel to verify operability upon loss of normal AC electrical power. The inspectors observed portions of ongoing testing, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. Two TDs and one PER were identified during the ongoing testing. TD-1 involved the CO2 Tank Low message for Step 6.2.6.2, Ensure all Software Errors and all Circuit Faults are corrected. WO 05-712990-00 was initiated to resolve the problem. TD-2 involved the trouble indications and alarms on local panels in that they were mis-programed. The indications were re-programmed by the vendor and the test was continued. PER 102140 was initiated to document TD-2. The inspectors determined that the testing successfully fulfilled the testing requirements for Stage 4A. There were no test exceptions.

1-PMTI-51368-STG04B, Rev. 0

This PMTI satisfied the post modifications test requirements for Stage 4B of DCN 51368. This stage involved modification activities to local fire control panel 0-LPNL-25-555. The test consisted of panel initial inspection, panel checkout, initial panel power up, panel programming, and circuit testing. The test performed was similar to testing for Stage 4A. The inspectors reviewed the completed test package and verified acceptance criteria for the test were satisfied. One TD was identified which involved a smoke detector fault and was similar to TD-1 of Stage 4A in that Step 6.2.6.2, Ensure all Software Errors and all Circuit Faults are corrected, was impacted by this condition. WO, 05-712991-00, was initiated to trouble shoot the item, the trouble shooting was successful, and the test was continued. The inspectors determined that the testing successfully fulfilled the testing requirements for Stage 4B. There were no test exceptions.

1-PMTI-51368-STG04C, Rev. 0

This PMTI satisfied the post modifications test requirements for Stage 4C of DCN 51368. The stage involved modification activities to the local fire control panel 0-LPNL-25-556. The test consisted of initial panel inspection panel, checkout, initial power up, panel programming, and circuit testing. Testing performed was similar to the test for Stage 4A. The inspectors reviewed the completed test package and verified acceptance criteria for the test were satisfied. Two TDs and two PERs were identified. TD-1 involved a smoke detector fault and was similar to TD-1 of Stage 4A in that Step 6.2.6.2, Ensure all Software Errors and all Circuit Faults are corrected, was impacted by this condition. WO 05-712989-00 was initiated to trouble shoot the item; the trouble shooting was successful and the test was continued. TD-2 involved the stopping of the 24-hour battery test on panel 0-LPNL-25-556 when the display on panel 0-LPNL-25-555 went blank. The display panel was re-programmed by the vendor and the test was continued. PER 101984 was initiated to document TD-1 and PER 102140 was initiated to document TD-2. The inspectors determined that the testing successfully fulfilled the testing requirements for Stage 4C. There were no test exceptions.

1-PMTI-51368-STG04D, Rev. 0

This PMTI satisfied the post modifications test requirements for Stage 4D of DCN 51368. This stage involved modification activities to the local fire control panel 3-LPNL-25-544. The test consisted of initial panel inspection, checkout, initial power up, panel programming, and circuit testing. Testing performed was similar to the test for Stage 4A. The inspectors reviewed the completed test package and verified acceptance criteria for the test were satisfied. One TD was identified which involved a heat detector fault and was similar to TD-1 of Stage 4A and Stage 4B in that Step 6.2.6.2, Ensure all Software Errors and all Circuit Faults are corrected, was impacted by this condition. WO 06-715944-00 was initiated to correct the problem and the test was continued. The inspectors determined that the testing successfully fulfilled the testing requirements for Stage 4D. There were no test exceptions.

b.3 Surveillance Instructions Used for Restart Testing Activities for Fire Protection Systems

Recently drafted Surveillance Instructions (SI) were also used to satisfy the post modifications test requirements for various local fire control local panels. The inspectors attended various pre-job briefs and post-job discussions. The inspectors also attended various meetings where testing activities, test planning, testing status, test exceptions, and test results were discussed. The inspectors observed portions of the ongoing testing, reviewed selected completed test packages, and verified acceptance criteria for testing were satisfied. A listing of specific SIs observed or reviewed are included in the attachment to this report.

c. Conclusions

Implementation of restart testing activities was acceptable. Only minor deficiencies which did not effect the results of the testing, were identified during performance of testing. Licensee processes were effective at identifying problems before components were placed in service.

Based on the above review and observations, the inspectors determined that testing was conducted according to applicable licensee procedures and emergent issues during the testing were adequately addressed by the licensee.

E1.6 Special Program Activities - Cable Installation and Cable Separation (37550, 37551)

a. Inspection Scope

The programs for investigating and resolving the issues of cable installation and cable separation are described in TVA's letter to the NRC dated May 10, 1991. This letter describes programs as essentially the same as described in the Browns Ferry Nuclear Performance Plan which outlined the corrective actions to be implemented before restart of Unit 2, and repeated for restart of Unit 3. NRC Inspection Manual Chapter 2509, Browns Ferry Unit 1 Restart Project Inspection Program, endorses the licensee Special Programs utilized on Units 2 and 3 as sufficient to address corresponding issues on Unit 1 if implemented in the same manner.

This inspection focused on violation NCV 259-2006-12-01, identified on a previous inspection, and the cable separations concern for Unit 1 Restart. This inspection included a review of the licensee's resolutions for PER 101868, DCN 51090 Stages 83, 84 Separations and PER 102752, Extent of Condition for PER 101868. In addition, the inspectors also selected additional examples of external and internal separations issues addressed in calculations EDQ 0999-910078 for external separation, EDQA 19992-003061 Internal Cable Separation Analysis for internal separation, and the criteria, methods, and exceptions identified in BFN 50-728, DCD Physical Independence of Electrical Systems.

The inspection was conducted by reviewing work order records, design basis documents, corrective actions, exceptions, drawings, and conducting walkdown inspections of methods used for achieving divisional separation or functional redundancy for Unit 1.

b. Observations and Reviews

b.1 Observation of Cable Installation Activities

The inspectors reviewed procedures for drawing control, walkdown control, plant modifications and engineering change controls to verify that licensee and contractor documented work practices were consistent with licensee commitments. BFN-50-728, Physical Independence of Electrical Systems, was reviewed to verify design requirements for the physical and electrical separation of electrical equipment and circuits for those systems whose operation is essential for the safe shutdown and isolation of the reactor. In addition, the inspectors selected examples of documents, used to make changes to DCNs in the field to verify that amendments appropriately captured modifications and were periodically added to permanent records. The inspectors selected 1-PNLA-009-003, 1-PNLA-009-030, 1-PNLA-009-033, and 1-PNLA-009-082 located in the Control Room and Auxiliary Instrument Room to verify that document control was consistent with design base documents for achieving divisional separation or functional redundancy. Per TVA's program for developing DCNs, the inspectors verified that design engineers were performing walkdowns or that waivers were approved by the appropriate personnel. The inspectors also reviewed licensee requirements for drawing control to assess design control practices that maintain the as-constructed state of the plant. The inspectors noted that the wiring diagrams illustrated physical dimensions of panels such as general layout, front, side, and top views, but did not describe the location of electrical equipment within panels. The inspectors noted that dimensional information present on some wiring diagrams could be incorrectly interpreted as representing physical state for individuals reviewing the drawings. This matter was discussed with the licensee, who indicated that the revised drawings would be annotated in a manner that would be obvious that the drawings are not physically representative of the as-constructed plant. The inspectors noted minor housekeeping issues in the Unit 1 Control Room panels concerning the controls used to protect the material condition of cables inside of the panels. The inspectors also observed training for cable splicing Raychem and TYCO type splices.

The inspectors identified no significant examples in which the corrective action program has not been effective at identification and resolution of issues related to cable separation issues. In this regard, the corrective action program has been effective. The inspectors reviewed the root cause, extent of condition, and corrective actions implemented by TVA related to PER 101868. The inspectors reviewed the justification related to PER 104357 which involved considering the effects of fire, ampacity, and cable tray fill. The inspectors also performed walkdowns related to PER102329, Cable Installation on Improper Side.

On a previous inspection, the inspectors identified NCV 05000259/2006012-02 involving separation of cables in panel 1-9-23 bay 8 in the control room. The inspectors found that the as-designed configuration did not match the as-constructed drawing configuration. The inspectors evaluated the root cause for PER 101868 and extent of condition for PER 102752 and found that the methods used were comprehensive. The inspectors verified that the scope for the extent of condition was comprised of electrical separations and Units 2 and 3 were evaluated for similar discrepancies. The inspectors did note weaknesses in the corrective actions for PER 101868 that responsible managers acknowledged and committed to bolstering. The inspectors reviewed PER 104357 and the justification for V3 and V4 cables being mixed in cable tray 35CA/2VDA on elevation 639' and going through the floor to elevation 593' through penetration R15935137. The inspectors reviewed section 5.2.4.3 of 50-758, Power, Control, and Signal Cables for Use in Class 1 Structures, for requirements related to mixing cables in cable trays and interviewed licensee engineers responsible for design deviations. The inspectors performed a walkdown of the area and verified that the justification performed was conservative and consistent with industry standards. Also, the inspectors reviewed preliminary progress for the trend analysis/commonality evaluation of configuration control that determines if configuration control weaknesses have been resolved.

b.2 <u>Review of Special Program Activities</u>

<u>Sidewall Pressure</u>

During a previous inspection, IR 2004-009, the Sidewall Pressure sub-program consisted of reviewing the details of the issue and the relevant design criteria. The inspectors independently reviewed the analysis and performed walkdowns of examples. The inspection of this sub-program consisted of reviewing calculations, cables, and physical arrangements. Calculation EDQ1 999 2003 0015, Analysis of Unit 1 Cable Installation - Miscellaneous Issues, Rev. 1 and Rev. 2, contains an evaluation of Unit 1 safety related cables in conduits, which may have experienced the following cable installation issues: excessive sidewall pressure, damage due to pullbys, jamming problems, and pulls through ninety degree condulets or mid-run flex conduit. During the previous inspection, the inspectors verified the calculation was acceptable and observed that a very limited number of originally installed cables required a pulling tension calculation. This was primarily because many cables are being replaced prior to restart and many power distribution cables had already been addressed under the Units 2 and 3 restart programs. During the same inspection, the inspectors also selected cables from an exception to verify that the licensee’s action was conservative for deviating from the acceptance criteria.

<u>Cable Pullbys</u>

During a previous inspection, IR 2004-009, the Cable Pullbys sub-program consisted of reviewing the details of the issue and the relevant design criteria. Calculation EDQ1 999 2003 0015, Analysis of Unit 1 Cable Installation - Miscellaneous Issues, was reviewed for this applicable subprogram to verify that the licensee had conservatively addressed the issue and that installation practices were consistent with industry standards. The inspection consisted of reviewing licensee methodology for performing cable pullbys and verifying field inspections of selected DCNs. In addition, the inspectors reviewed a sample of calculations.

Cable Jamming

During a previous inspection, documented in IR 2004-007, inspection of the Cable Jamming subprogram consisted of reviewing the details of the issue, the licensee's proposed corrective actions, and the implementation of modifications. Calculation EDQ1 999 2003 0015, Analysis of Unit 1 Cable Installation - Miscellaneous Issues, was reviewed for this applicable subprogram. The inspection consisted of reviewing licensee methodology for identifying cables that could be damaged by jamming during the pulling-in process.

Vertical Cable Supports

During a previous inspection, documented in IR 2004-009, the Vertical Cable Supports sub-program consisted of a detailed review of the licensee's methodology and a walkdown inspection of the Unit 1 control complex to look for examples of this issue. Calculation EDQ1 999 2003 0016, Analysis of Cable Support in Vertical Raceway for Unit 1, was reviewed to verify that safety-related cables, installed in vertical raceways and exceeding the cable support spacing as specified in G-38, were analyzed. The previous inspection consisted of reviewing licensee methodology for installing cable supports, a selection of conduits, and performing quality control verification of walkdown data and dispositions.

c. Conclusions

Cable installation activities continued to be performed in accordance with documented requirements. Additionally, based on current and previous reviews, the inspectors determined that implementation of four sub-programs for the Cable Installation Special Program were proceeding in accordance with licensee commitments and regulatory requirements. These sub-programs include sidewall pressure, cable pullbys, cable jamming, and vertical cable supports. Completed actions to address these issues for Unit 1 are consistent with those previously committed to and performed for Units 2 and 3. The inspectors concluded that no issues related to these sub-programs that would negatively impact the restart of Unit 1 were identified as the result of the above reviews. No further inspections are anticipated for these four sub-programs. However, implementation activities associated with cable separation and the one remaining cable installation sub-program, bend radius of medium voltage cables, will need further inspections by the NRC to verify corrective actions are in accordance with licensee commitments.

E1.7 Special Program Activities - Q-List (37550)

a. Inspection Scope

The inspectors evaluated various ongoing activities associated with the licensee's Q-List Special Program. The inspectors reviewed the licensee's program and procedures to determine adequacy of component safety significance determinations. Additionally, inspectors reviewed the licensee's implementation of required Q-List updates as part of the ongoing NRC review of system return to service activities.

Inspectors reviewed licensee procedures SPP-9.6, Master Equipment List (MEL); NEDP-4, Q-List and UNID Control; and 0-TI-414, Component Labeling, Signs, Operator Aids, and Permanent Information Postings. The inspectors also reviewed MEL package tracking, update, and input processes for two systems that have been returned to Operations under the SRTS: System Pre-operability Checklist II (SPOC) program. The systems reviewed were System 23 (RHRSW) and System 67 (EECW).

b. Observations and Findings

10 CFR 50, Appendix B, Criteria II requires that all safety-related structures, systems, and components (SSCs) be identified. TVA implements this requirement by use of the Q-List. On January 9, 1991, TVA submitted their plan for implementation of the Q-List. The Q-List for Unit 2 was developed following recovery of that unit and the Q-Lists for Units 1 and 3 were to be developed prior to recovery of those units. Implementation of the Unit 3 Q-List was previously reviewed by the NRC as documented in Inspection Report 50-259,260,296/95-43. During that review the inspectors concluded that the Unit 3 Q-List development was in accordance with regulatory requirements and commitments. The inspectors determined that the Q-List, as developed for Unit 1, was actually a report available from a larger electronic database. This database is the MEL and consists of a comprehensive database of component information for safety-related and nonsafety-related SSCs. The inspectors determined that the licensee's program required that all newly installed components and components touched by a design change were to have a determination of safety significance. Additionally, all components which were outside the physical scope of design changes (retained components) were required to be reviewed for safety significance determination. The Q-List report consists of a list of all components with a valid safety classification in MEL (safety related field marked yes). MEL update packages are developed as part of the design change process and become part of the DCN package. MEL package update data entries are processed through the licensee's Enterprise Maintenance and Planning Control (EMPAC) system. The licensee informed the inspectors that a goal had been established to complete all MEL updates within 30 days following receipt of an update package.

The inspectors reviewed Self Assessment BFR-RSU-04-001, MEL Data. This self assessment was performed by a team of Unit 1 managers and engineers during October and November 2003. The self assessment was performed to determine the effectiveness of Unit 1 MEL Program activities. During this assessment the licensee concluded that the program was meeting program requirements. However, several minor deficiencies related to management expectations, training, and untimely MEL updates were identified.

The inspectors also reviewed several completed Nuclear Assurance (NA) observations performed on MEL activities during the last year. These NA observations were associated with MEL updates for design changes for the RHR and RHRSW systems. The specific NA observation reports and PERs reviewed are listed in the Attachment to this inspection report. During the performance of those observations the licensee NA assessor reviewed various DCN MEL update packages, evaluated tracking of MEL packages from package creation until completion of updates, and verified safety classification for selected components (modified and retained components). Additionally, the assessor spot checked DCN MEL update requests to verify the classification information was consistent with Unit 3 component classifications. In most cases program requirements and management expectations were being satisfied. However, one minor example of an incorrect EMPAC status for an RHR flow transmitter was identified. The associated MEL package update had been completed but the individual component status was shown as unverified. PER 96459 was issued to document this problem.

The inspectors determined that only about 16% of the expected updates had been completed as of June 2006. Previous NRC inspections of the licensee's SRTS process had included specific reviews of information from EMPAC for two risk significant systems which had recently completed SPOC II acceptance by the site operating organization. Specifically, the inspectors reviewed selected DCN MEL update packages to determine the adequacy of MEL updates for the RHRSW and EECW Systems. The inspectors reviewed RHRSW and EECW system MEL packages and tracking status. The inspectors verified completion statuses were accurate and that all system DCNs requiring component additions, deletions, or modifications were documented. A ten percent sample (approximately eighty components) of component unit identifiers (UNIDs) were verified accurate using MEL package requested data input and Enterprise Maintenance and Planning Control (EMPAC) database verification. The sample included both modifications and non-modification maintenance and labeling. In addition, fifteen components were physically walked down and inspected in the plant. The inspectors verified physical component characteristics, location, labeling, and select procurement data. Two PERs were reviewed and applicable corrective actions verified.

c. Conclusions

The inspectors determined that activities associated with MEL data updates for the RHRSW and EECW systems were adequately input to the existing EMPAC database and that the database accurately represented components installed in the field. Based on the above review the inspectors determined that the licensee's program for development of the Unit 1 Q-List was being performed in accordance with the documented requirements. Completed or planned actions were consistent with those previously performed for Units 2 and 3. However only a small portion of the expected MEL updates have been completed; those updates will continue to be reviewed by the inspectors as part of future reviews of the SRTS process. No further inspections of this Special Program are anticipated. No violations or deviations were identified.

E1.8 Special Program Activities - Small Bore Piping and Instrument Tubing (37550)

a. Inspection Scope

The small bore piping (less than 2.5 inch diameter) program was developed by the licensee to address concerns identified with application of design criteria, incomplete support details, questions regarding seismic qualification, and lack of design calculations. The small bore piping includes instrument tubing, but does not include piping which had been rigorously analyzed, such as the CRD piping. The licensee's program to resolve the concerns involves identification of the small bore piping and instrument tubing systems; performance of walkdown inspections to identify inadequately supported piping and tubing, missing supports, and missing hardware from existing supports; preparation of as-built drawings; completion of design calculations to qualify the small bore piping and tubing; issuing DCNs to correct discrepancies; and implementation of the DCNs.

The licensee's commitments for resolution of issues associated with the small bore piping and instrument tubing are documented in TVA letter dated December 13, 2002, Subject: Browns Ferry Nuclear Plant - Unit 1 - Regulatory Framework for the Restart of Unit 1. The letter references previous commitments for restart of Units 1 and 3 stated in a letter dated July 10, 1991, Subject: Regulatory Framework for the Restart of Units 1 and 3, and NRC approval of the licensee's plans in a letter dated April 1, 1992. Design criteria for design and seismic qualification were submitted to NRC in the following TVA letters: Subject: Action Plan to Disposition Concerns Related to Units 1 and 3 Small Bore Piping, dated February 27, 1991; Subject: Action Plan to Disposition Concerns Related to Units 1 and 3 Instrument Tubing, dated February 27, 1991; and Subject: Small Bore Piping, Tubing, and Conduit Support Plan for Units 1 and 3 - Additional Information, dated December 12, 1991. Acceptance of the licensee's program for resolution of the small bore piping and instrument tubing concerns by NRC is documented in Safety Evaluation Reports dated October 24, 1989 and January 23, 1991. Previous NRC inspections of the small bore pipe support program are documented in Inspection Report numbers 50-259/2005-008 and 50-259/2006-006.

b. Observations and Findings

The inspectors reviewed walkdown procedures and design criteria, reviewed results of walkdown inspections performed by licensee engineering personnel, reviewed design calculations and DCNs, walked down selected small bore piping and instrument tubing systems, and examined completed modifications. These systems included System 71, Reactor Core Isolation Cooling (RCIC); System 64, Primary Containment Isolation; and portions of System 85, CRD, piping which were not rigorously analyzed.

The inspectors reviewed results of walkdown inspections, design calculations, design change documents, and examined completed modifications. The inspectors reviewed calculations which evaluated the deficiencies and the design output documents (DCNs) which specified the required field work to correct the deficiencies.

The inspectors walked down portions of the instrumentation tubing and small bore piping listed below to verify that the design changes were implemented in accordance with the design documents. Attributes examined were support location, configuration, including member size and type, weld size, hardware for attachment of piping/tubing to supports, and support attachment to building structure. The inspectors also examined supports which were identified with missing or incorrect hardware to verify the correct type hardware was installed as specified in the DCN design drawings.

Supports examined were as follows:

- System 71 (RCIC) support numbers: 1-47B452–2060-01 through -2060, -2068, -2069, -2070, -2072, -2074, -2075, -2076, -3297, 0-47B36-66, 1-47B456-2064-01 through -2064-04, and 1-47B456-2064-07 through -18, and 1-47B456-2064-43, -2064-44

- System 64 (Primary Containment Isolation) support numbers: 1-47B600-5416-36, 1-47B600-5416-37, 1-47B600-5416-43, and 1-47B600-5438 through 1-47B600-5442

- System 85 (CRD) support number 1-47B466-31

During examination of support number 1-47B600-5440 on the instrumentation for primary containment isolation system, the inspectors identified that hardware item number 2 installed on the support was a Unistrut channel N5000 (3 1/4" in height) versus the Unistrut channel N1000 (1 5/8" in height) specified by the design drawing. Installation of the apparently incorrect hardware item was inspected and accepted by Quality Control (QC) inspectors.

10 CFR 50, Appendix B, Criteria V, Instructions, Procedures, and Drawings, in part, states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The

design drawing for support 1-47B600-5440 specified the use of Unistrut channel N1000 for hardware item number 2. Contrary to the above, the inspectors identified that hardware item number 2 installed on the support was a Unistrut channel N5000. The failure to construct support number 1-47B600-5440 in accordance with design drawing requirements was identified to be a violation. This violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy and will be identified as the first of two examples of Severity Level IV NCV 50-259/2006-007-01, Failure to Construct Instrument Tubing Supports in Accordance with Design Drawings.

This issue was documented in PER 100642, Installation of Incorrect Unistrut Hardware on Instrument Line Support. Design engineering analyzed the as-constructed support and determined it was acceptable.

The inspectors also identified several discrepancies between completed supports and the approved design drawings when examining supports on the RCIC system. The licensee provided copies of an engineering change control document, Post Issuance Change (PIC) 65455, which authorized the changes. Final review of PICs are being conducted by engineering. The licensee stated that the drawings will be revised to incorporate the approved PICs. In the case where a PIC is not approved, a work order will be issued to perform additional field work to make any changes necessary so the affected supports meet design criteria.

c. Conclusions

During the walkdown inspection, the inspectors verified the following attributes complied with the requirements shown on the design drawings: support locations, support member sizes and configuration, weld sizes, type, and length, connection details, and verification of correct type of hardware for attachment of small bore piping/tubing to supports. One of two examples of an NCV of 10 CFR 50, Appendix B, Criterion V was identified for failure to construct instrument tubing supports in accordance with design drawings requirements.

The inspectors determined that the licensee’s program for correction of deficiencies identified in support of small bore piping and instrument tubing complies with the design criteria, commitments to NRC, and NRC requirements. The inspectors had previously determined that small bore piping and instrument tubing installed inside the drywell were acceptable. However, additional samples of small bore piping and instrument tubing installed in the reactor building (outside the drywell) will be inspected prior to closure of this Special Program.

E1.9 Special Program Activities - Instrument Sensing Lines (37550)

a. Inspection Scope

The instrument line program was developed for restart of Browns Ferry Unit 2 to address issues regarding installation of instrument sensing lines. The issues concerned potential violations of three basic design requirements: physical separation of redundant components; provision of sensing line slope; and specification of material quality requirements. TVA submitted the corrective action program for Unit 2 to address these concerns in a letter dated August 14, 1989. The Unit 2 scope and common instrument scope were based on evaluations of system calculations, the FSAR Chapter 14 safety analysis, emergency operating instructions, review of instrument related maintenance problems, and the master component equipment list. TVA concluded after completion of the instrument line evaluations that problems were limited to instrument slope. No cases were identified of inadequate physical separation of redundant components, and no cases were identified of inadequate material quality. The Unit 2 instrument sensing line program was reviewed and approved by the NRC, as documented in Section 3.4 of NUREG-1232, Volume 3, Supplement 2. In letters dated February 13 and November 8, 1991, TVA submitted their action plan to resolve concerns related to instrument sensing lines for Browns Ferry Units 1 and 3. The basis approach was to use the same methodology used for Unit 2. NRC accepted the TVA action plan in a Safety Evaluation Report dated December 10, 1991.

Eliminated from this Special Program were vendor supplied instruments, instruments with a process pressure greater the 100 psig, totally sealed capillary tubing, and instruments without sensing lines.

b. Observations and Findings

The inspectors reviewed walkdown procedures and design criteria and reviewed results of walkdown inspections performed by licensee engineering personnel. The inspectors also examined DCN 51177 which specifies corrective actions to address instrument sensing line slope deficiencies on the RHRSW line and walked down the RHRSW instrument sensing lines to examine completed modifications. In addition, the inspectors reviewed a sample design calculation which will serve as a basis to document that adequate instrument line slope exists and that process root valves are installed with proper stem orientation.

The inspectors walked down portions of the RHRSW instrumentation tubing in the Unit 1 reactor building between the RTVs installed in the RHRSW lines at Elevation 583' - 9" to Instrument Panel 25-62 on Elevation 524' to verify that the design changes were implemented in accordance with the design documents. The inspectors independently measured instrument line slope at numerous locations using a level engineer plumber to verify the modified instrument tubing met the slope specified on the DCN drawings. The inspectors also reviewed work order and quality control inspection records documenting correction of the slope deficiencies identified during the licensee's initial walkdown inspections.

The inspectors examined new instrument line tubing supports installed to support the RHRSW instrument sensing lines. The following supports were inspected: numbers 1-47B600-4887 through 4893, 4895, 4896, 4897, 4903 through 4907, 4909, 4910, 4912, 4913, 4915, and 5735. Attributes examined were support location, configuration, including member size and type, weld size, and hardware for attachment of piping/tubing to supports, and support attachment to building structure installed as specified in the DCN design drawings. Acceptance criteria utilized by the inspector were specified in Modification and Addition Instruction MAI-4.2A, Piping/Tubing Supports, Rev. 19 and MAI-4.4a, Instrument Line Installation, Rev. 16.

During examination of RHRSW instrument tubing support numbers 1-47B600 - 04905 and 1-47B600 - 04910, the inspectors identified that incorrect clamps were installed to attach the instrument tubing to the support structure. The clamps installed were Unistrut part N1112 clamps with 10 gauge guide plates which provide two way restraint. The installation drawings, drawing numbers 1-47B600 - 04905 and 1-47B600 - 4910, specify Unistrut N1111 clamps, which provide three way restraint of the tubing. The supports had been inspected and accepted by quality control inspectors. The system had been accepted and turned over to Operations. This issue was documented in PER 102320. Design engineering re-analyzed the as-constructed instrument line and determined that the stresses in the tubing were acceptable. The new tubing stress analysis showed the loads on seven supports increased due to the change in configuration of the installed tubing. However the new support loads did not exceed the design allowable limits. The licensee determined that two different quality control inspectors had inspected and accepted the support/tubing installation with the incorrect clamps.

10 CFR 50, Appendix B, Criteria V, Instructions, Procedures, and Drawings, in part, states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Drawing numbers 1-47B600-4905 and 1-47B600-4910 specify Unistrut N1111 clamps for attachment of instrument tubing to support numbers 1-47B600-4905 and 1-47B600-4910 Contrary to the above, on May 3, 2006, the inspectors identified that the instrument tubing was attached to the supports using Unistrut N1112 clamps with spacer plates. This resulted in a change in the configuration of the instrument tubing system which invalidated the tubing stress analysis and support design. The incorrect clamps had been inspected and accepted by QC inspectors. This violation is an additional example of NCV 50-259/2006-007-01, Failure to Construct Instrument Tubing Supports in Accordance with Design Drawings.

The inspectors reviewed the corrective actions associated with PER 06-102320 which the licensee initiated on May 3, 2006, to document and disposition this issue. The corrective actions included a review to determine extent of condition and cause, re-inspection of additional supports, review of quality control inspection documentation, and a review of the corrective action data base to identify other similar events. The licensee subsequently installed the correct clamps on the supports as part of the corrective actions under PER 102320. The inspectors re-examined support numbers 1-47B600-4905 and 1-47B600-4910 to verify the correct clamps were installed.

c. Conclusions

During the walkdown inspection, the inspectors examined the following attributes and compared the instrument lines and installed supports with the requirements shown on the design drawings: instrument line slope, support locations, support member sizes and configuration, weld sizes, type, and length, connection details, and verification of correct type of hardware for attachment of instrument tubing to supports. A second example of an NCV of 10 CFR 50, Appendix B, Criterion V was identified for failure to construct instrument tubing supports in accordance with design drawings requirements.

The inspectors determined that the licensee's program for correction of instrument tubing slope issues complies with the design criteria, commitments to NRC, and NRC requirements. With the exception of the two examples identified in the NCV, instrument line supports were installed in accordance with the design drawing requirements. However, additional samples of instrument tubing installed in Unit 1 will be inspected prior to closure of this Special Program. No findings of significance were identified.

E1.10 Special Program Activities - Large Bore Piping and Supports Program (50090)

a. Inspection Scope

The inspectors reviewed BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7. The inspectors selected and performed independent walkdown inspections for nine pipe supports in Reactor Water Cleanup, Reactor Heat Removal, and Re-Circulation systems to verify the field installed conditions as compared to as-built drawings. The inspectors selected one large bore support calculation in the Re-Circulation system for review. The inspectors reviewed applied load directions such as axial, vertical, or horizontal, from isometric drawings in the stress calculations to compare the load directions provided in final as-built pipe support drawings. The inspectors verified the current scope of the Large Bore Program for IE Bulletin 79-14 included the required piping boundary by reviewing the Residual Heat Removal system isometric drawings. The inspectors reviewed two DCN and WO Packages in order to verify the adequacy of the design or modification, inspection, and implementation for the pipe supports. The inspectors reviewed PERs to verify adequacy of problem identification, resolution, corrective actions, and extent of condition review. The independent support walkdown and calculation review were preferred to verify adequacy and compliance with the design criteria, drawings, IE Bulletin 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchors, and IE Bulletin 79-14, Seismic Analysis for As-Built Safety-Related Piping Systems.

b. Observations and Findings

The inspectors walked down nine supports with licensee QC examiners and engineers. The inspections were performed to evaluate the effectiveness of the licensee's walkdown, modifications, and repairs. The elements inspected included dimensions, sizes, diameters, symbols, identifications, spacing, and clearances for members, anchor bolts, base plates, standard components, and welds. The supports walked down are listed below:

Support No.	Drawing No. & Revisions
1-47B406-285	1-47B406-285, Rev. 003
1-47B406-287	1-47B406-287, Sheets 1 & 2, Rev. 002 &005
1-47B406-290	1-47B406-290, Rev. 001
1-47B452-3037	1-47B452-3037, Rev. 003
1-47B465-464	1-47B465-464, Rev. 002
1-47B465-500	1-47B465-500, Sheets 1 & 2, Rev. 004 &002
1-47B465-501	1-47B465-501, Rev. 001
1-47B465-507	1-47B465-507, Rev. 002
1-47B465-546	1-47B465-546, Sheets 1 & 2, Rev. 004 &002

Two minor discrepancies were identified, which the licensee stated to be a draft change error and a detail cut out from a wrong item. PERs 104574 and 104573 respectively, were issued by the licensee to correct the drawings.

The inspectors reviewed support design calculation CDQ1-068-2002-0216, Rev. 003, for support 1-47B465-546. The elements in the support calculation reviewed included assumptions, design methodology, special requirements or limitations, computer model, computer design input data, computer output data, computations and analyses, summary of results, conclusions, and attachments. The computer input data included node numbers and coordination, member numbers, end nodes, and properties, joint fix, member releases, seismic coefficient, loads and load combinations, weld sizes and configurations, base plates, anchor bolts, pipe support load transmittal from the stress group, structural attachment loading schedule to Civil Engineering Group, and allowable stresses for the members.

The inspectors reviewed Piping System 23 (DCN 51177), Residual Heat Removal and Piping System 78 (DCN 51351), and Fuel Pool Cooling to verify that all the pipe supports and load directions provided in the final as-built drawings met the required support types and load directions specified in the isometric drawings contained in the pipe stress calculations.

The inspectors selected a piping layout drawing for review, Piping System 74, Seismic Class 1 Boundary - RHR System, from calculation ND-Q0999-920011, Rev. 42, Seismic Class 1 System Piping Boundary, which indicated the boundary for safety related piping. The inspectors reviewed the isometric drawings contained in the stress calculations or stress problems N1-174-1RA, -2R to -5R, -7R, -9R, -11R, -14R, -15R, -17R, -18R, -27R, and 28R to verify that the stress isometric drawings contained the boundary scope identified in the piping layout drawing for the RHR system and that the required safety related piping was included in IE Bulletin 79-14 requirements.

The inspectors reviewed DCN 51028 and WO 02-009379 for System 75 Core Spray and DCN 51347 and WO 03-008315 for System 74 RHR. The inspectors reviewed the DCNs for the scope, design, modification, or repair drawings, 50.59 screening review, procedures or calculations required to be revised or generated, and test or inspection requirements. The inspectors reviewed WOs for installation, material, welder and weld data records, inspection records, and nondestructive examination records such as magnetic particle, liquid penetrant, or visual examination records.

The inspectors also reviewed several PERs contained in the DCNs or WOs for their root cause analysis, evaluation, disposition, corrective actions, and extent of condition review.

c. Conclusions

Based on independent walkdowns of nine pipe supports, one pipe support calculation, as-built support drawings; and problem resolution the inspectors determined that licensee performance was adequate in the Large Bore Pipe Support Special Program. However, additional samples will need to be inspected prior to closure of this Special Program.

E1.11 Special Program Activities - Fuse Program (51053)

a. Inspection Scope

In Section III.13.6 of the BFN Nuclear Performance Plan, Rev. 2, TVA described corrective actions for an electrical problem involving the misapplication of fuses that limit current in overload protection. The corrective action program as it was applied to support Units 2 and 3 restart contained the following actions:

- Revise the BFN fuse substitution program control document to reflect the appropriate standards.

- Perform calculations using revised design standards to specify the appropriate fuses for each application and document this activity on the fuse tabulation document.

- Conduct plant walkdowns to determine and document the installed fuses for compliance with the fuse tabulation, with the exception of motor control centers, where allowable substitution has been identified.

- Compare the results of the fuse tabulation with the walkdown for reconciliation.

- Document and resolve by the corrective action process all inadequate fuses identified.

- Delete and replace fuse ratings on design drawings with a fuse identification before restart. The fuse tabulation would be the single source of fuse requirements for the applicable fuses.

This inspection examined the fuse program activities that were being implemented for restart of Unit 1. The inspectors reviewed fuse sizing calculations, design drawings, problem evaluation reports, and performed field walkdown inspections of a selected sample of breaker cubicles to assess the adequacy of the licensee's implementation of the fuse special program.

b. Observations and Findings

The inspectors examined the installed fuses in a random sample of ten Motor Control Center (MCC) breaker cubicles to verify that new replacement fuses were being installed in accordance with the Unit 1 Fuse Program. The specific cubicles and components examined are listed in the Attachment to the report. The inspectors conducted field inspections and compared the installed fuse nameplate data against the Master Equipment List, design drawings, and fuse sizing calculations to verify that the installed fuses were the correct size and type as specified by design. The inspectors also verified that the unverified assumptions used as inputs in design calculations were being properly dispositioned by design.

The inspectors requested a data query of the licensee's corrective action program files to identify those Problem Evaluation Reports with the key word "fuse" included in the text. The search covered the period from December 27, 2004 to June 24, 2006. A summary report of the PERs identified by the search was provided to the inspectors for review. The inspectors reviewed the report and selected several PERs associated with fuse problems to determine if the licensee had implemented adequate corrective actions for the issues. A list of the PERs reviewed are included in the Attachment to the report.

c. Conclusions

The inspectors concluded that the licensee's corrective action program to resolve the problems with misapplication of current-limiting fuses is acceptable to support Unit 1 restart. The program is equivalent in scope to those previously applied to the restart of the other units at Browns Ferry. The inspectors examined a select sample of the replacement fuses and confirmed that the program was being adequately implemented. Therefore, no further inspection of this Special Program is anticipated.

E1.12 Special Program Activities - Electrical Cable Splices and Terminations in Equipment Qualification Applications (51053)

a. Inspection Scope

In 1986, the NRC issued Information Notice 86-53 alerting licensees to a potential safety problem involving improper installation of heat-shrinkable tubing over electrical splices and terminations. In addition to this information notice, an employee concern was raised at BFN regarding problems with existing site procedures for installing electrical splices. Based on these concerns, TVA initiated a comprehensive program at BFN to ensure the adequacy of all class 1E electrical cable splices and terminations in harsh environments.

TVA's comprehensive splice program as described in the Nuclear Performance Plan, Rev. 2, required all splices and terminations subject to 10 CFR 50.49 to be inspected and replaced if the splices did not meet installation standards. This program was implemented as part of the restart effort on Units 2 and 3. The NRC staff reviewed the implementation of this program during the restart of Units 2 and 3 and found it to be acceptable.

Unlike Units 2 and 3, the Unit 1 Restart cable splice program required fewer inspections of existing cable splices since most of the splices were being replaced prior to restart of Unit 1. There are approximately 533 total Unit 1 EQ splices. Of those, 515 new EQ splices will be installed as part of the Unit 1 restart. The remaining 18 EQ cable splices that are not being replaced were verified through walkdown inspections to be in compliance with the splice installation standards.

During this inspection, the inspectors reviewed the licensee's on-going activities to implement the cable splice program for Unit 1 restart. The inspectors reviewed completed splice work order records, problem evaluation reports, conducted walkdown inspections of selected EQ splices, witnessed two in-process cable splices, and attended portions of a splice training class to assess the adequacy of the licensee's implementation of the EQ cable splice special program.

b. Observations and Findings

The inspectors conducted field inspections of three as-built heat shrink cable splices and witnessed two in-process heat shrink cable splices to assess the adequacy of the licensee's implementation of the EQ cable splice and termination special program. The three completed cable splices (1-$R-023-1661A, 1-$R-023-1686A, and 1-$PC-069-0305A) were randomly selected from a list of newly completed EQ cables splices. The two in-process cable splices were being worked in accordance with the licensee's modification completion schedule. The inspectors reviewed the associated work order records and performed field inspections of the as-built splices to verify that the splice configurations as well as the materials used to assemble the splices were in accordance with the manufacturer's instructions and plant modification procedures. The inspectors noted that the as-built configuration of splice 1-$R-023-1686A was not consistent with the manufacturer's instructions because the breakout boot was not fully covered by the outer sleeve. The legs of the breakout extended past the outer sleeve. However, this configuration was later determined to be acceptable based on additional information obtained from the vendor and reviewed by the inspectors.

The inspectors also witnessed two heat shrink splice terminations being made by the craft. The first splice was a butt splice on pigtail wires in a conduit seal assembly with in-line heat shrink outer sleeves. The splices were associated with the termination of flow solenoid valve 1-FSV-75-58. The inspectors observed that the wires were stripped using the appropriate tool(s); the butt splice connectors on each of two wires were properly crimped; the wire insulation was adequately cleaned prior to applying heat shrink material; and based on vendor use range tables, the correct heat shrink material was installed to make the splice.

The second splice installation witnessed by the inspectors was a high voltage motor connection kit in-line type splice at the 4 KV 1B Core Spray pump motor. The inspectors observed the splice being made on the C-phase of the motor pigtail cable. The splices on the other two phases were similar but they were not observed. The inspectors observed that the craft and QC followed the step by step instructions from the splice kit to make the splice. The inspectors also observed that QC was present during the installation of both splices.

Additionally, the inspectors observed ongoing work activities associated with electrical terminations of two solenoid valves on System 85 CRD performed under DCN 51240-1, Pull and Terminate New EQ Cables, and in accordance with Work Order 03-006607-027. The inspectors verified splicing activities associated with Flow Solenoid Valve 1-FSV-85-37A, CRD Scram Discharge Volume Drain and Vent Pilot Valve A, and 1-FSV-85-35A, CRD Back-Up Scram Pilot Valve A, were conducted in accordance with the MAI and work order requirements. The terminated cables were 1RP203-IIIA and 1RP206-IIIA, respectively. The inspectors specifically reviewed Heat Shrink Installation Data Sheets (DS-6) and verified data entered as required. The inspectors also verified craft installer, verifier, and supervision training certification.

In addition to the above, the inspectors requested a data query of the licensee’s corrective action program files to identify those PERs with the key word “splice” included in the text. The search covered the period from December 27, 2004 to June 24, 2006. A summary report of the PERs identified by the search was provided to the inspectors for review. The inspectors reviewed the report and selected several PERs associated with cable splice problems to determine if the licensee had implemented adequate corrective actions for the identified splice issues. A list of the PERs reviewed are included in the attachment to the report.

c. Conclusions

Ongoing activities associated with the electrical terminations using Raychem splices were conducted in accordance with existing requirements. No violations or deviations were identified. Additionally, the inspectors concluded that the special program for electrical cable splices and terminations in EQ applications was adequate to support Unit 1 restart. The program will replace most of the EQ splices on Unit 1. The inspectors confirmed, by examination of a select sample of completed splices and by witnessing two in-process splices, that the program was being adequately implemented. Therefore, no further inspection of this Special Program is anticipated.

E1.13 Special Program Activities - Thermal Overloads (51053)

a. Inspection Scope

In Section III.13.4 of the BFN Nuclear Performance Plan, Rev. 2, TVA described a design control problem with the application of thermal overloads (TOLs) in 480 Volt alternating current (ac) and 250 V direct current (dc) motor control centers. The corrective action program as it was applied to support Units 2 and 3 restart contained the following actions:

- Inspect the 480V ac and 250V dc safety-related motor control centers to determine and document the installed TOL ratings.
- Develop and issue a sizing criteria for TOLs.
- Evaluate the walkdown results against the sizing criteria.
- Replace or reset improperly sized TOL elements, as appropriate.
- Properly sized or replaced TOLs will be documented on a TVA design drawing to assure that current and future installations of thermal overloads are correct.
- For those Unit 2 harsh environment safe shutdown TOLs with qualification deficiencies, TVA will issue a design to disable the TOLs by disconnecting the control circuit interlocks until qualified TOLs are obtained.

This inspection focused on the corrective actions that were being implemented by TVA to resolve the thermal overload concern for Unit 1 Restart. This inspection was conducted by reviewing design drawings, design calculations, and conducting walkdown inspections of as-built thermal overload installations and end devices. The inspectors also reviewed the licensee’s corrective actions for NCV 50-259/2005-08-02, Measures Were Not Adequate to Assure That The TOLs in 480V MOV Board 1B Cubicles 14C-2 and 15C Were Strapped Out.

b. Observations and Findings

The specific components examined are listed in the Attachment.

The inspectors examined the installed thermal overload (TOLs) heaters in a random sample of ten MCC breaker cubicles to verify that the correct TOLs were installed in accordance with design basis documents (i.e., TOL sizing calculations and design drawings). The inspectors found all ten MCC cubicles to have the correct TOL or jumper installed as specified by design documents. In addition to the above, the inspectors walked down the end devices (i.e., MOVs and continuous duty motors) and recorded nameplate motor data (e.g., horsepower rating, service factor, starting current, and full load amps.) to compare with values used as inputs in the TOL sizing calculations. Some of the end devices were not installed or available for review by the inspectors.

Also, some of the non-safety motors were observed to be either smaller or larger than the values used in the calculations. The inspectors found that the licensee had assumed in the TOL sizing calculations that some of the Unit 1 non-safety motors were the same as those in Units 2 and 3. However, this assumption was later proved to be wrong because some of the Unit 1 motors were different from the other units. For example, the Unit 1 Drywell Equipment Drain Pump Motors 1A and 1B were assumed in the TOL sizing calculation to be the same as Units 2 and 3 with a motor full load current of 4.8 amps. However, the Unit 1 motors had a full load current less than Units 2 and 3 of 4.1 amps. This assumption resulted in some of the heaters being slightly oversized; however, the motors would still be protected with the installed TOLs. The inspectors also noted that the installed non-safety Unit 1 HPCI Gland Seal Condenser Blower Pump motor was larger than that assumed in the TOL sizing calculation. The motor had been replaced with a larger motor without a design change being issued. The installed motor was a 2/3 horsepower (HP) motor versus a 1/3 horsepower motor assumed in the calculation. The thermal overload heater installed was undersized for the 2/3 HP motor and it would have tripped in approximately 2.8 seconds. It is likely this problem would have been identified by the licensee during system testing. The licensee documented the problem in the corrective action program as PER 106704. The inspectors concluded that the problems with non-safety motors were not violations of NRC requirements. The inspectors did confirm that the nameplate data for safety-related motors examined were in accordance with the TOL sizing calculations.

The inspectors reviewed the licensee's corrective actions for the non-cited violation associated with the failure to jumper out the TOLs on certain safety-related MOVs. The inspectors reviewed the licensee's evaluation of the root causes and extent of condition for the violation. The inspectors found that the licensee utilized a combination of events and causal factors as well as barrier analysis methodologies to determine the root cause and extent of causes for the violation. The licensee determined that a lack of guidance within the work procedures led to the failure to incorporate a PIC to strap out the TOL that had been initiated and issued by design engineering. The licensee determined that the extent of condition was limited to the TOLs that were to be strapped out as part of the Unit 1 restart effort. The inspectors verified that work orders had been implemented to jumper out the required TOLs.

c. Conclusions

The inspectors concluded that the licensee's program to resolve problems with sizing of thermal overloads is acceptable to support Unit 1 restart. The licensee has replaced or strapped out the Unit 1 safety-related thermal overloads in accordance with the plant design. Therefore, no further inspection of this Special Program is anticipated.

E1.14 Special Program Activities - Moderate Energy Line Break (37550)

a. Inspection Scope

The inspector reviewed the licensee's actions associated with the Moderate Energy Line Break (MELB) Recovery Special Program and the subsequent flooding recovery actions. This review included a review of the initial MELB submittals and TVA's commitment to review the effects of flooding due to breaks in moderate energy lines outside primary containment. The inspector also reviewed Nureg-1232, Volume 3, Supplement 2, Browns Ferry Unit 2 "Safety Evaluation Report on Tennessee Valley Authority: Browns Ferry Nuclear Performance Plan" for Unit 2 and W87 040615 001, Moderate Energy Line Break (MELB) Flood Evaluation Report for Browns Ferry Nuclear Plant Unit 1 Extended Power Uprate, Rev. 1, dated June 11, 2004. The inspectors also reviewed plant procedures, training materials, and operator training for response to MELB events.

b. Observations and Findings

TVA's program to establish MELB protection, and an integrated response to MELB, were documented in MELB Flood Evaluation Report for Browns Ferry Nuclear Plant Unit 1 Extended Power Uprate, Rev. 1. This report documented the investigation into the flooding effects of breaks postulated to occur in moderate energy piping systems that were routed throughout Browns Ferry Unit 1 and common Class I buildings and structures. This report contained an analysis of the effects of extended power uprate. The inspectors compared these results with the results for the MELB reports for Units 2 and 3. The inspectors concluded that TVA had demonstrated MELB will not adversely impact the ability of the licensee to achieve and maintain Unit 1 shutdown.

The inspectors reviewed the original flood studies requested by the Atomic Energy Commission (AEC). The areas of the plant included in this evaluation were the Reactor Building, the Intake Pumping Station, the Diesel Generator Building, the Turbine Building, the Service Building, the Radwaste Building, the Office Gas Building, and the Stack. The inspectors reviewed the assumptions and verified the June 11, 2004 analysis included the significant differences from the original AEC flood studies and the material provided for the Unit 2 and 3 MELB reports and found that any differences were appropriately addressed.

The inspector also reviewed materials provided to operators in both initial and requalification training. The materials provided appropriate entry conditions, appropriate diagnostic tools, and appropriate mitigation strategies. This included appropriate instructions to achieve and maintain a shutdown condition.

Finally, in a letter from NRC to TVA dated June 7, 2006, the NRC staff stated they had reviewed the Unit 2 MELB and found it acceptable as documented in a Safety Evaluation Report (SER) dated January 23, 1991. The letter further stated "Since Unit 1 uses the same methodology, NRC follow-up will consist of a Region II inspection on the Unit 1 MELB program implementation. This item will be updated on the Recovery Issues Checklist, as no Chief of Nuclear Reactor Regulations safety evaluation is required." This inspections is the documentation of the Unit 1 MELB implementation inspection described in the NRC letter.

c. Conclusions

Based on observations, documents reviewed, and discussions with training personnel, the inspectors determined that TVA had adequately address MELB. The inspectors reviewed the NRC to TVA letter dated June 7, 2006 and confirmed that the licensee had adequately implemented the MELB Special Program.

No violations or deviations were identified during the review of the licensee's MELB Special Program. Based on the results from this inspection, the inspector concluded that the MELB Special Program was adequate. Therefore, no further inspection of this Special Program is anticipated.

E1.15 Emergency Equipment Cooling Water Flow Testing (37550)

a. Inspection Scope

The inspectors reviewed the Heat Sink (HS) program activities, which included Unit 1, to ensure activities were in compliance with licensee commitments and NRC requirements, and to determine the potential readiness to transition future Unit 1 inspections of HS performance to the Reactor Oversight Process (ROP). The ROP is the NRC's inspection program for operating reactors, and selected inspection areas (designated as Cornerstones) of the ROP can be incorporated for Unit 1 once NRC inspections conclude that the area can be adequately monitored under the ROP. The transition process is described in NRC Manual Chapter 2509.

b. Observations and Findings

The inspectors reviewed site and corporate heat exchanger (HX) program procedures, GL 89-13 commitment letters, HX inspection and cleaning work orders, and frequency of inspection for selected Unit 1 HXs to verify these were in accordance with licensee commitments. The inspectors also reviewed specific cleaning procedures and work orders to verify this had been updated to include Unit 1 HXs. In addition, the inspectors reviewed minimum flow testing acceptance criteria, updated Unit 1 flow testing procedures, and Unit 1 flow trends to verify HS systems were being maintained in accordance with design basis requirements.

The inspectors also reviewed general health of the Unit 1 EECW and RHRSW systems via review of design basis documents, intake structure diver inspections, updated corrosion monitoring procedures, corrosion coupon monitoring trends, raw water program strategic plans, WOs for Unit 1 dead leg flushes, and discussions with the EECW system engineer.

c. Conclusions

The licensee's HS program was being adequately maintained. All changes to procedures and to the program were being performed in accordance with licensee commitments and NRC requirements. Based on focused HS reviews for Unit 1, inspectors did not identify any impediments to the future transition of Unit 1 HS inspections under the Initiating Events and Mitigating Systems Cornerstones to the Reactor Oversight Process.

E1.16 Inservice Inspection Activities

a. Inspection Scope

The inspectors reviewed the Browns Ferry Unit 1 Inservice/Preservice Inspection (ISI/PSI) activities as detailed below to ensure these activities were in compliance with regulatory requirements and licensee commitments. See NRC Inspection Reports 50-259/2004-007, 50-259/2004-009, 50-259/2005-006, and 50-259/2005-008 for previous inspections in this area.

As detailed in the licensee's ISI program, the first ten-year ISI interval, which began August 1, 1974 for Browns Ferry Unit 1, is currently in its third period and will end one year following the restart of the unit. The applicable codes for the ISI/PSI programs are ASME Section XI, 1995 Edition, 1996 Addenda, and for sample selection ASME Section XI 1974 Edition with Addenda through Summer 1975.

b. Observation/Review of ISI/PSI Activities

The inspectors conducted an on-site review of nondestructive examination (NDE) activities to evaluate compliance with Technical Specifications and the applicable editions of ASME Sections III, V, IX and XI to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of ASME acceptance standards.

Specifically, the inspectors observed the following examinations:

Manual Ultrasonic Examination (UT)

- Weld Number RWCU-1-S001-007, Reactor Water Clean-Up Elbow to Pipe, ASME Class 1

Liquid Penetrant (PT) Examination

- Weld Number RWCU-1-S001-007, Reactor Water Clean-Up Elbow to Pipe, ASME Class 1

Specifically, the inspectors reviewed the following examination records:

PT Examination

- Weld Number 1-47B465-512-1A, Reactor Water Recirculation, Pipe to Pipe, PSI, ASME Class 1
- Weld Number 1-47B465-508-1A, Reactor Water Recirculation, Pipe to Pipe, PSI, ASME Class 1

UT Examination

- Weld Number CS-1-009.002, Core Spray, Elbow to Pipe, PSI, ASME Class 2
- Weld Number RCH-1-4V, Reactor Pressure Vessel (RPV) Top Head Meridonal Weld, ISI, ASME Class 1
- Weld Number N2J Nozzle, Nozzle to RPV Shell, ISI, ASME Class 1

Magnetic Particle (MT) Examination

- Weld Number FCV-1-52, Main Steam Isolation Valve, PSI, ASME Class 1

Visual Examination (VT)

- Support Number 1-47B415-47, Feedwater Pipe Support, PSI

Specifically, the inspectors reviewed the following examination records that contained recordable indications:

- NOI (Notice of Indication) U1C6R-026, Main Steam FCV-1-38 Bonnet Studs and Nuts, ASME Class 1
- NOI U1C6R-028, DWLNR-1-3.45, Dry Well Liner Thickness Measurements
- NOI U1C6R-031, Feedwater N4A Subsurface Indication, Nozzle to Shell Weld, ASME Class 1

For the above examinations, inspectors reviewed the examination data sheets, equipment calibration records, examination procedures, and examination personnel certifications. The PSI/ISI inspection activities and records were compared to the applicable requirements.

The inspectors performed a review of ISI and Welding Program related problems that were identified by the licensee and entered into their corrective action program. The inspectors reviewed a sample of these corrective action documents to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

c. Conclusions

The inspectors determined that the licensee’s ISI and PSI activities met applicable code requirements and licensing commitments. Changes to the program since the last inspection were consistent with licensee commitments and NRC requirements. Based on this review and past reviews of the Unit 1 ISI program, inspectors did not identify any impediments to the planned transition of Unit 1 ISI inspections under the Initiating Events, Barrier Integrity, and Mitigating Systems Cornerstones to the ROP.

E1.17 Replacement of Floating Head Assemblies for Residual Heat Removal Heat Exchangers 1A and 1C (71111.08, 37551)

a. Inspection Scope

The inspectors reviewed the selected ASME Section XI Repair/Replacement activities for Unit 1 RHR Heat Exchangers.

b. Observations and Findings

The inspectors reviewed the completed ASME Section XI Repair/Replacement packages associated with installation of new floating head assemblies for RHR Heat Exchangers 1A and 1C. Replacement of the floating heads was not part of the original Unit 1 Recovery work scope. However, the heat exchangers for all three units have experienced degradation and Heat Exchanger 1C experienced significant leakage which prevented completion of all planned startup testing as scheduled. The licensee procured 12 replacement floating heads to allow refurbishment of heat exchangers for all three units. Heat Exchangers 1A and 1C on Unit 1 were selected as the first candidates for replacement.

The inspectors reviewed WOs 04-724879-011 and 04-724879-012 which documented the installation of the new floating heads, split ring and seal gasket, and channel head bolting for RHR Heat Exchangers 1A and 1C. These components were replaced on these heat exchangers as part of the licensee's program to refurbish the RHR heat exchangers for all three units (12 heat exchangers).

During the above review the inspectors noted that the licensee had not performed hydrostatic testing following the replacement of the floating heads. Based on discussions with licensee engineering personnel and review of the applicable ASME Code requirements, the inspectors determined that hydrostatic testing of the heat exchangers was not required. Code Case N–416-3 allowed the licensee to perform a system leakage test with visual examination (VT-2) at nominal operating pressure and temperature in accordance with IWA-5000 of the 1992 Edition of ASME Section XI rather than hydrostatic testing.

c. Conclusions

Based on a review of selected ASME Section XI Repair/Replacement activities, the inspectors concluded that the licensee repair activities were performed in accordance with the documented requirements.

E8 Miscellaneous Engineering Issues (92701)

E8.1 (Closed) Generic Letter 88-11, Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations.

GL 88-11 referenced Rev. 2 to regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Material. It points out that the NRC does not accept the owners group position that the margins given by following the procedures of Appendix G or 10 CFR 50 can be safely reduced. On January 8, 1993, the NRC issued Amendment 190 to the Browns Ferry Unit 1 Facility Operating License to address radiation embrittlement of reactor vessel and its impact on plant operations. This TS amendment was granted after TVA responded to a request for additional information (TVA letters dated October 24, 1991 and July 19, 1991). These letters contained revised pressure/temperature limit curves as described in GL 88-11. The NRC staff found these curves acceptable. This meets the requirements of GL 88-11. This issue is closed for Unit 1.

E8.2 (Closed) Generic Letter 92-01, Reactor Vessel Structural Integrity, Revision 1 and Revision, 1 Supplement 1

GL 92-01 Rev. 1 references compliance with requirements and commitments regarding reactor vessel integrity. Revision 1, Supplement 1, required that TVA collect and report new data pertinent to analysis of structural integrity of their reactor pressure vessel and to assess the impact of that data on their reactor pressure vessel integrity analysis relative to the requirements of 10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50 Appendices G and H. TVA submitted the required data and the required analysis (TVA letters dated July 7, 1992, December 1, 1992, and August 2, 1993. In a letter to TVA from the NRC dated April 19, 1994, the NRC stated that TVA had provided the requested data. On June 10, 1998, the NRC request additional information on weld chemistry data. TVA responded in a letter dated September 8, 1998 that current pressure-temperature curves and other corresponding information submitted in support of GL 92-01 were based on conservative weld chemistry data and did not require revision. They submitted their analysis with this letter which supports this position. The NRC has not requested additional information. The inspectors reviewed historical documents submitted and concluded that TVA has submitted all the information required by GL 92-01. This issue is closed for Unit 1.

E8.3 (Closed) GL 83-28, Required Actions Based on Generic Implications of Salem ATWS Event

GL 83-28 requested licensees to respond to certain generic concerns that developed from the investigation into the failure of scram circuit breakers at the Salem Nuclear Power Plant. These concerns were categorized into four areas: post trip review, equipment classification and vendor interface, post maintenance testing, and reactor trip system reliability improvements. The inspectors reviewed licensee response letters to the GL, NRC Requests for Additional Information letters, and NRC Safety Evaluations (see Attachment) regarding GL 83-28.

The inspectors found that most of the required actions of this GL had been previously addressed by safety evaluations. Remaining items, which included Items 1.2, 4.5.2, and 4.5.3, were tracked pending closure by licensee commitments.

The inspector found that Item 4.5.2 and 4.5.3 of the GL identified an NRC position that required on-line functional testing of the reactor trip system, including independent testing of the diverse trip features. NRC Safety Evaluation dated September 2, 1986 documented a conclusion that TVA had demonstrated a sufficient basis for not requiring modification of the backup scram function to provide for on-line testing (Item 4.5.2). The Safety Evaluation concluded that the backup scram function should be tested during refueling outages, and that such testing should be included in plant technical specifications. The inspector verified that testing of the backup scram function is required by TS 3.3.1.1.14, Reactor Protection System Instrumentation Logic System Functional Testing with a frequency corresponding to refueling outages. The inspector verified that this testing was being performed for Units 2 and 3 and that the licensee planned to include this testing in Unit 1 Surveillance Procedure 1-SR-3.3.1.1.12, Reactor Protection System Mode Switch in Shutdown Scram and Logic System Functional Test. NRC Safety Evaluation dated August 17, 1990 concluded that the existing intervals for on-line functional testing (Item 4.5.3) at Browns Ferry Unit 2 were consistent with achieving high reactor trip system availability based on TVA endorsement and application of Boiling Water Reactor Owner's Group (BWROG) Reports NEDC-30844 and 30851P. All licensing actions for Item 4.5.3 were considered complete for Units 1, 2, and 3.

NRC Safety Evaluation dated June 12, 1985 concluded the response to Item 1.2, Data and Information Capability, was acceptable. This was based on plans to install an upgraded process computer and an enhanced sequence of events recorder. By letter, dated November 9, 1993, TVA informed the NRC of the completion of the commitment to upgrade the hardware for Unit 2. Due to the capability of the hardware actually installed, the functions of the sequence of events and time history recorder were also accomplished by the process computer. The inspector verified that the licensee planned to install hardware on Unit 1 with the same capability using DCN 51082.

Various NRC Multiplant Action (MPA) Items and Unresolved Safety Issues (USI) which relate to this GL had been previously addressed by the licensee as part of the recovery of Units 2 and 3. TI 2515/95 (MPA C-002), Inspection for Verification of Boiling Water Reactor (BWR) Recirculation (Recirc) Pump Trip; TI 2500/20, Inspection to Determine Compliance with ATWS Rule, 10 CFR 50.62; and USI A-075 (MPA B-085), Salem ATWS Item 1.2 Post-Trip Data Review and Information Capability, were previously reviewed by the NRC for Unit 3 restart and considered acceptable. This review was documented in Inspection Report 50-259,260,296/95-60. These areas were addressed as part of the licensee's resolution of GL 83-28 for Unit 1 recovery. Additionally, the inspector determined that the licensee was implementing the same designs with the same level of redundancy and diversity for the Unit 1 ATWS, alternate rod insertion, and Recirc Pump Trip functions as had previous been implemented for Units 2 and 3.

The inspectors determined that no further actions were required for Unit 1. Therefore, because these modifications are being tracked under the facility modification process and any deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.4 (Closed) IE Bulletin (IEB) 79-12 Short Period Scrams at BWR Facilities

The inspectors reviewed IEB 79-12, Short Period Scram at BWR Facilities. The issue concerned high flux detected by the intermediate range (IRM) neutron monitors during an approach to criticality. The bulletin was intended to reduce the number of challenges to the Reactor Protection System (RPS) high flux scram. The bulletin required BWR facilities to: review and revise operating procedures to ensure an estimate of critical rod pattern prior to each approach to criticality, verify procedures require notch step withdrawal before the estimated critical position is reached where inaccuracies in critical rod pattern are anticipated, assure that rod withdrawal sequences minimize notch worth of individual rods, evaluate operability of emergency rod in switch, describe reactor operator training program regarding the four prior items, and provide a written response to the NRC. Licensee actions associated with this bulletin for Unit 3 was previous reviewed and closed in NRC inspection report 50-259,260,296/95-51. This report describes TVA's NRC-approved banked withdrawal sequences and reduced notch worth startup process. Unit 1 Technical Specification limiting conditions of operation (LCOs) 3.6.1and 3.3.2.1 and surveillance requirements (SR) 3.1.6.1 and SRs 3.3.2.1.2, 3.3.2.1.3, 3.3.2.1.5 and 3.3.2.1.7 control the withdrawal sequence and reduced notch worth. The Unit 1 version of procedure 1-GOI-100-1A, Unit Startup and Power Operation, was not yet available for review. However, the inspectors reviewed Unit 2 procedure 2-GOI-100-1A and verified that it incorporated reduced notch worth and banked position withdrawal sequences. Since Unit 1 was implementing similar actions as used on Units 2 and 3, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.5 (Closed) IEB 88-07, Power Oscillation in BWRs

IEB 88-07 requested that all holders of operating licenses of BWRs take actions to ensure that adequate operating procedures, instrumentation, and operator training is provided to prevent the occurrence of uncontrolled power oscillations during all modes of BWR operation. The bulletin was written following a BWR dual recirculation pump trip event in 1988 after which the unit experienced an excessive neutron flux oscillation while the unit was on natural circulation. Supplement 1 to IEB 88-07 was issued to provide additional information concerning power oscillations in BWRs. The supplement requested that each licensee take action to ensure that the safety limit for the plant minimum critical power ratio is not violated. TVA responded and confirmed actions required by the bulletin were completed by letter dated November 4, 1988. TVA responded to Supplement 1 and confirmed plans to implement the interim stability recommendations by letter dated March 6, 1989. The proposed TS changes to implement the reactor core thermal-hydraulic stability recommendations were provided by TVA letter dated January 14, 1992 and approved by the NRC in a letter to TVA dated May 31, 1994. The inspectors reviewed procedures 2-GOI-100-1A, 2-SR-3.3.1.1.I, 2-

AOI-68-1A, and 2-AOI-68-1B and verified that the procedures implemented the required actions in the bulletin and supplement. TVA initiated a PER for an instance where an operating crew in training did not manually scram the plant prior to the automatic action occurring during a loss of recirc pump trip with power oscillations scenario. The inspectors reviewed PER 102236 and determined that the corrective actions were adequate to address the problem description. At the time of this inspection, BFN Unit 1 had not yet implemented the digital upgrade of the neutron monitoring system or completed the procedure revisions similar to Units 2 and 3. Additionally BFN Unit 1 had not yet implemented the Oscillation Power Range Monitor (OPRM) Option III similar to Units 2 and 3. BFN Unit 1 Power Range Neutron Monitor Upgrade with Implementation of Average Power Range Monitor and Rod Block Monitor was submitted with TS change 430 dated November 10, 2003. BFN Unit 1 OPRM was submitted with TS change 443 dated January 6, 2006. BFN Unit 1 plans to startup with its extended power uprate and the power range neutron monitor digital upgrade and oscillation power range monitor implemented. Since Unit 1 was implementing similar actions as used on Units 2 and 3, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.6 (Closed) GL 94-02, Long-Term Solutions and Upgrade of Interim Operating Recommendations For Thermal Hydraulic Instabilities in BWRs

The inspectors reviewed GL 94-02, Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in BWRs. The GL requested licensees to take appropriate actions to augment their respective procedures and training for preventing or responding to thermal-hydraulic instabilities in the reactor and to submit a plan describing the long-term stability solution option selected and the implementation schedule for modification of the plant protection systems to ensure compliance with General Design Criteria 10 and 12 of 10 CFR 50 Appendix A. This GL along with IEB 88-07 and NRC Temporary Instruction (TI) 2515/099, BWR Power Oscillations, all concerned BWR hydraulic instabilities that can be experienced in areas of high power and low recirculation flow. By letter dated September 8, 1994, TVA provided its response to the requested actions of GL 94-02. By letter dated December 22, 1994, TVA notified the NRC that the training program upgrades, procedure revisions, and necessary training requested by GL 94-02 were completed November 25, 1994 for Unit 2. By letter dated October 4, 1995, TVA provided the installation schedule for the Stability Long-Term Solution for NRC GL 94-02. By letter dated May 24, 1999, TVA notified the NRC that the Unit 2 Oscillation Power Range Monitor (OPRM) module of the digital power range neutron monitor was satisfactorily installed, tested, and enabled completing the stability solution for Unit 2. By letter dated May 25, 2000, TVA notified NRC that the Unit 3 OPRM module was satisfactorily installed tested and enabled completing the stability solution for Unit 3. At the time of this inspection, Browns Ferry Unit 1 had not yet implemented the digital upgrade of the neutron monitoring system or completed the procedure revisions similar to Units 2 and 3. Additionally, Unit 1 had not yet implemented the OPRM Option III similar to Units 2 and 3. BFN Unit 1 Power Range Neutron Monitor Upgrade with Implementation of Average Power Range Monitor and Rod Block Monitor was submitted with TS change 430 dated November 10, 2003. Unit 1 OPRM was submitted with TS change 443 dated January 6, 2006. Unit 1 plans to startup with its extended power uprate and the power range neutron monitor digital upgrade and oscillation power range monitor implemented. Since Unit 1 was implementing similar actions as used on Units 2 and 3, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.7 (Closed) TMI Action Item I.D.1, Control Room Design Review, NUREG 0737

As a result of the NRC Three Mile Island (TMI) Action Plan outlined in NUREG-0660, operating reactor licensees and applicants for operating reactor licenses were required to perform a detailed Control Room Design Review (CRDR) to identify and correct design discrepancies. In accordance with Supplement 1 to NUREG-0737, which confirmed and clarified the CRDR requirement in NUREG-0660, each licensee was required to conduct its CRDR on a schedule negotiated with the NRC.

TVA submitted a generic program plan for a CRDR of each of the TVA nuclear facilities. An effort was conducted for Browns Ferry Nuclear Plant Units 1, 2 and 3 by TVA to address the nine CRDR requirements outlined in NUREG-0737. The NRC staff, assisted by two independent vendors, evaluated the licensee's review and documented the results in a Safety Evaluation Report (SER) dated August 9, 1988. The staff concluded that the licensee's review had not fully met all aspects of the nine requirements of NUREG-0737. The licensee addressed these issues in two supplemental responses dated November 9, 1988 and August 22, 1991. In SER dated October 29, 1991, the NRC staff concluded the detailed CRDR program at BFN had met all the requirements, and that any changes to the implementation schedule for Units 1, 2, and 3 must be submitted to the NRC for approval.

TVA notified the NRC staff of program completion for the CRDR for Units 2 and 3 in letters dated June 14, 1993 and February 9, 1996, respectively. The letters included the licensee's corrective action plan for the resolution of each safety significant CRDR Human Engineering Discrepancy (HED) identified through the initial audit. Among the HEDs described, some were common between all units, some common between two units, and others were unit specific. Unit 1 HEDs were to be resolved prior to restart.

The inspectors reviewed the original correspondence documenting the CRDR effort for Units 2 and 3. During the review, the inspectors identified any safety significant HEDs or other deficiencies, identified during the initial detailed CRDR audits, applicable to Unit 1. The inspectors selected a sample of high safety significant HEDs for Unit 1 to verify the licensee had developed modification design packages or other appropriate corrective actions to address and resolve the identified issues. The inspectors conducted a walkdown of the Unit 1 Main Control Room to observe and verify the accuracy of completed and implemented HED design modifications. Additionally, the inspectors observed different stages of other various modifications being implemented to resolve identified HEDs. The inspectors also discussed the plan for how HED design changes would be integrated into the Operator Training Program.

Therefore, because the licensee's original submittals were adequate, the licensee has effectively assessed and integrated high safety significant HEDs into the design modification program, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. The inspectors determined that no further actions were required for Unit 1. This issue is closed for Unit 1.

E8.8 (Closed) TMI Action I.D.2, Safety Parameter Display System, NUREG-0737, and GL 89-06, Safety Parameter Display System

The NRC staff issued NUREG-0737, which provided guidance for implementing Three Mile Island (TMI) action items. Generic Letter 82-33 transmitted Supplement 1 of NUREG-0737 to all licensees, to clarify the TMI action items related to Emergency Response Capability, including the Safety Parameter Display System (SPDS). A staff evaluation determined that a large percentage of licensee designs did not fulfill the requirements identified in Supplement 1 to NUREG-0737. The NRC staff issued Generic Letter 89-06 to require licensees to assess and certify their SPDS fully meet the requirements of Supplement 1, or certify that the SPDS will be modified to fully meet the requirements of NUREG-0737. If a certification could not be provided, the licensees were required to provide a discussion of the reasons and any intended compensatory actions. Enclosed with GL 89-06, was NUREG-1342, which documented the NRC staff's experience with SPDS implementation requirements and methods the staff found both acceptable and unacceptable.

In a letter dated January 14, 1985, TVA submitted an implementation schedule for the Browns Ferry units. A program plan was developed for Units 1, 2 and 3 by TVA to address the requirements outlined in NUREG-0737. TVA was requested by the NRC staff to provide additional, detailed information on the human factors program and hardware isolation features associated with the SPDS. In a TVA response letter dated December 19, 1989, the licensee described Phase 1 SPDS, which TVA committed to having functional prior to the Unit 2 restart. The letter discussed the basic operation of the Phase 1 SPDS, the human factors review, and verification and validation program. In correspondence dated December 11, 1990, TVA notified the NRC staff of implementation of the SPDS final design. The NRC staff identified several deficient areas regarding the final design and documented these issues in a SER dated March 6, 1991. The licensee's supplemental response to this SER, outlining corrective actions, was reviewed and the staff subsequently determined BFN fully met the SPDS design requirements of NUREG-0737, Supplement 1 as documented in SER dated February 5, 1992. TVA notified the NRC staff of program completion for the implementation of the Browns Ferry SPDS in a letter dated October 19, 1993. The SPDS for Unit 1 would be installed prior to restart of that unit.

The inspectors reviewed the original submittals documenting the SPDS implementation program. During the review, the inspectors identified the necessary critical safety functions to be monitored and any commitments that may have been made applicable to Unit 1. The inspectors discussed with the licensee the hardware and software changes that were being made and any identified impact on the current SPDS system installed on the operating units. Additionally, the inspectors reviewed the design modification package for implementing the new Integrated Computer System (ICS) to verify appropriate system parameters and indications were identified. The inspectors determined that no further actions were required for Unit 1. Therefore, because the licensee's revised design submittal was adequate, the licensee has effectively addressed the SPDS requirements in the current design modification program, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.9 (Closed) GL 89-13, Service Water System Problems Affecting Safety-Related Equipment

On July 18, 1989, the NRC issued GL 89-13, Service Water System Problems Affecting Safety-Related Equipment. This GL requested licensees to supply information about their respective service water systems, and to follow recommended or equally effective actions to ensure compliance of the service water systems with 10 CFR Part 50, Appendix A, General Design Criteria 44, 45, 46 and Appendix B, Section XI. On April 4, 1990, the NRC issued Supplement 1 to GL 89-13 to document additional information discussed during workshop sessions held by the NRC to discuss GL 89-13 with licensees.

By letter dated March 16, 1990, the licensee submitted their response to GL 89-13, with subsequent letters providing updates and changes to BFN's original response submitted on December 31, 1990 and August 17, 1995.

The licensee's response letter to GL 89-13 included a discussion of the service water systems at Browns Ferry and a response to each of the NRC recommended actions specified in the GL.

In response to GL 89-13, the licensee specified Residual Heat Removal Service Water (RHRSW) and Emergency Equipment Cooling Water (EECW) as the systems in scope for the GL. In response to the GL five recommended actions, the licensee described the existing programs used to address raw water service concerns and discussed additional commitments. The licensee's program included actions to inspect the intake pump pits and to establish chemical treatments and corrosion monitoring programs for both systems. In addition, the licensee committed to test RHRSW and EECW pumps to verify design flow, to measure and trend differential pressure across the RHR heat exchangers (HX), and between the RHRSW pump discharge and the RHR HX inlet for flow blockage. The licensee also committed to inspect, clean, and to flow test safety-related components in the EECW system to verify flow requirements. Piping 4 inches or less in diameter were replaced with stainless steel, and the licensee committed to inspect portions of the RHRSW and EECW systems when opened for preventive maintenance.

In January, 2006, the inspectors verified that Browns Ferry had implemented the GL 89-13 commitments and existing programs into the Unit 1 procedures, preventive maintenance programs for the RHRSW system, and partially for the EECW system, to assure compliance. The purpose of this inspection was to complete the verification that the licensee had incorporated GL 89-13 commitments for the Unit 1 EECW system. That NRC review was documented in Inspection Report 50-259/2005-09.

To finish the inspection for the EECW system, the inspectors reviewed updated EECW corrosion monitoring trends and procedures. In addition, the inspectors reviewed heat sink program, fouling control, chemical treatment, and flushing procedures to verify procedures had been updated to include Unit 1. The inspectors also reviewed preventive maintenance tickets, model work orders, and frequency of inspections for Unit 1 EECW HX inspections to verify they were in accordance with GL 89-13 requirements. Updated flow verification procedures and trends were also reviewed to verify they were consistent with design basis requirements.

The inspectors determined that the actions and programs in place at Browns Ferry prevent flow blockage, component degradation, and corrosion issues for the Unit 1 EECW and RHRSW systems were similar to the Unit 2 and 3 solutions with the same process, and would effectively address the GL 89-13 commitments. Therefore, this item meets the closure criteria established for Unit 1 recovery issues and is closed for Unit 1.

E8.10 <u>(Closed) GL 88-01, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping</u>

GL 88-01 requested licensees to provide their plans for replacement, inspection, repair, and leakage detection of piping susceptible to IGSCC, and state whether they intend to follow the NRC staff positions or propose alternatives. This item was previously reviewed by NRC as documented in Inspection Report 50-259/06-06. Subsequently, the Office of Nuclear Reactor Regulation completed their review in this area as documented in SER issued on May 30, 2006. Based on that review, the staff concluded that the licensee’s supplemental response to GL 88-01 for Unit 1 was in compliance with the staff’s position on IGSCC in BWR austenitic stainless steel piping. The inspectors determined that no further actions were required for Unit 1. Therefore, because this item is effectively being tracked in the licensee’s corrective action program, is being corrected similarly to the Unit 2 and 3 solutions with the same process, and because any implementation deficiencies would likely be detected by the licensee’s oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.11 <u>(Closed) Inspector Followup Item (IFI) 50-259/05-09-02, Reactor Pressure Vessel (RPV) Lower Head Deposits</u>

During inspection of the RPV internals area below the core plate, the licensee identified numerous corrosion deposits. These deposits were located in the lower head region on the upper side of numerous CRD stub tubes between the tube and the sloping vessel wall. A total of 53 of 185 stub tube locations were identified to have these deposits. Based on discussions with General Electric personnel, the licensee concluded that these types of deposits had been previously identified at other BWRs and have not been known to cause any operational or safety issues. The licensee had selected four of these areas for removal of the corrosion deposit, chemical analysis of deposit material, and examination of cladding surface in the removal area. The licensee left the remaining deposits in place. This IFI was left open pending additional NRC review of the basis for the licensee's decision for not fully removing the corrosion deposits, and also for review of the licensee's actions associated with PER 92079 to verify that there would be no detrimental effects to the RPV or internals. The inspectors independently reviewed a sample of the licensee's enhanced visual examination (EVT) activities including the resolution demonstration of the inspection technique. The inspectors also reviewed examination and certification records for the inspection personnel. The licensee's EVT of the removal areas did not identify any pit or crack-like indications in the cladding. This exam was performed to verify that no potential flaws, which might have resulted from the presence of the deposits, existed in those areas and to verify that the removal process, which utilized mechanical chipping and hydrolazing, did not significantly damage the RPV cladding. Furthermore, the inspectors did not identify any additional concerns in their review of the licensee's evaluation to not remove the remaining deposits. The inspectors also communicated with an expert from NRC headquarters who indicated that these corrosion deposits would not be unexpected and historical operating experience at other BWRs has shown that where cladding cracks and gouges are present, they have never resulted in concern for the integrity of the RPV. Therefore, because this item is effectively being tracked in the licensee's corrective action program and because any implementation deficiencies would likely be detected by the licensee's oversight programs, and have only minor consequences, this item meets the closure criteria established for the Unit 1 recovery issues. This issue is closed for Unit 1.

E8.12 (Closed) Multiplant Action Item (MPA) B011, Flood of Equipment Important to Safety

The aspects of this item associated with restart of Unit 1 are covered by existing Recovery Special Program, Moderate Energy Line Break. Closure of that Special Program is documented in Section E1.14 of this inspection report. This item is redundant and no further NRC action is required for restart of Unit 1. This item is administratively closed.

E8.13 (Closed) MPA B041, Fire Protection - Final Technical Specifications

The aspects of this item associated with restart of Unit 1 are covered by existing Recovery Special Program, Fire Protection 10 CFR 50, Appendix R. This Special Program will be reviewed for closure prior to restart of Unit 1. This item is redundant and no further action is required for restart of Unit 1. This item is administratively closed.

E8.14 (Closed) Unresolved Safety Issue (USI) A-7, Mark 1 Long Term Program, NUREG 0661, Supplement 1

The aspects of this item associated with restart of Unit 1 are covered by existing Recovery Special Program, Long Term Torus Integrity Program. This Special Program will be reviewed for closure prior to restart of Unit 1. This item is redundant and is considered administratively closed.

E8.15 (Closed) USI A-9, ATWS

The aspects of this item associated with restart of Unit 1 are covered by GL 83-28. Closure of that GL is documented in Section E8.3 of this inspection report. This item is redundant and no further NRC action is required for restart of Unit 1. This item is administratively closed.

E8.16 (Closed) USI A-24, Qualification of Class 1E Safety-Related Equipment

The aspects of this item associated with restart of Unit 1 are covered by existing Recovery Special Program, Environmental Qualification and Recovery Special Program, Component and Piece Parts Qualification. These Special Programs will be reviewed for closure prior to restart of Unit 1. This item is redundant and no further action is required for restart of Unit 1. This item is administratively closed.

E8.17 (Closed) USI A-26, Reactor Vessel Pressure Transient Protection

The aspects of this item associated with restart of unit 1 were covered by the review of Generic Letter 88-11. Closure of that GL is documented in Section E8.1 of this inspection report. This item is redundant and no further action is required for restart of Unit 1. This item is administratively closed.

III. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Program

a. Inspection Scope

The inspectors continued to observe and/or review ongoing licensee maintenance program activities. Maintenance work activities were controlled by approved procedures and work orders. Specific maintenance activities reviewed and observed included selected portions of ongoing activities associated with return to service of the RHRSW System, CRD System, Drywell and Emergency Control Air, and Containment Air Dilution (CAD) System.

b. Observations and Findings

Licensee maintenance activities reviewed or observed by the inspectors during this report period were associated with the return to service of the RHRSW System, CRD System, Drywell and Emergency Control Air, and CAD System. These activities included support for system testing. Specific maintenance activities reviewed or observed included the following:

- WO 03-001414-39 for the support of the pneumatic testing of Drywell and Emergency Control Air in the Reactor Building
- WO 03-006734-29 for the replacement of the packing for bypass valve 1-BYV-85-551 in CRD System in the Reactor Building
- WOs 03-019008-00 and 03-019010-00 for the disassembly and inspection of flow control valves 1-FCV-84-19 and 1-FCV-84-20, respectively, in the CAD System in the Reactor Building
- WO 06-710104-00 for the disassembly and inspection of check valve 1-CKV-32-655 in Drywell and Emergency Control Air System in the Reactor Building
- WO 03-017668-01 for the installation and testing of the new valve operator for flow control valve 1-FCV-85-83A in the CRD System in the Reactor Building
- WO 04-724879-12 for removal and replacement of the 1C RHR Heat Exchanger floating head and the overall work involved with the heat exchanger was documented in previous inspection reports. The reports documented that Eddy-Current Testing (ET) indicated a significant number of tubes were degraded. As a result of the ET the licensee decided to replace 450 tubes. At the end of this report period the licensee was in the process of replacing heat exchanger tubes.

The inspectors reviewed the applicable WO packages and observed selected portions of the ongoing maintenance activities. The inspectors determined that WO packages included sufficient guidance to allow maintenance personnel to adequately perform the associated work activity. Maintenance personnel and foreman were knowledgeable of applicable requirements and appropriately documented work actually performed, as required by plant procedures.

c. Conclusions

No deficiencies were identified during the review of the ongoing maintenance activities. The Maintenance organization continued to provide appropriate and comprehensive repairs to Unit 1 components which do not require design changes to support Unit 1 Restart. Maintenance WO packages included sufficient technical guidance to allow maintenance personnel to adequately perform the associated work activity. Maintenance personnel and foremen were knowledgeable of applicable requirements and appropriately documented work actually performed, as required by plant procedures.

V. Management Meetings

X1 Exit Meeting Summary

On August 1, 2006, the resident inspectors presented the inspection results to Mr. Masoud Bajestani and other members of his staff, who acknowledged the findings. Although some proprietary information may have been reviewed during the inspection, no proprietary information will be identified in the final inspection report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

M. Bajestani, Vice President, Unit 1 Restart
R. Baron, Nuclear Assurance Manager, Unit 1
M. Bennett, QC Manager, Unit 1
D. Burrell, Electrical Engineer, Unit 1
P. Byron, Licensing Engineer
J. Corey, Radiological and Chemistry Control Manager, Unit 1
W. Crouch, Nuclear Site Licensing & Industry Affairs Manager
R. Cutsinger, Civil/Structural Engineering Manager, Unit 1
B. Hargrove, Radcon Manager, Unit 1
K. Hess, SWEC Project Director
E. Hollins, Maintenance and Modifications Manager, Unit 1
R. Jackson, Bechtel
R. Jones, General Manager of Site Operations
S. Kane, Licensing Engineer
D. Kehoe, Nuclear Assurance, Unit 1
J. Lewis, Integration Manager
G. Little, Restart Manager, Unit 1
J. McCarthy, Licensing Supervisor, Unit 1
R. Moll, Mechanical Engineering and Systems Engineering Manager, Unit 1
J. Ownby, Project Support Manager, Unit 1
J. Schlessel, Maintenance Manager, Unit 1
J. Symonds, Modifications Manager, Unit 1
J. Valente, Engineering Manager, Unit 1

INSPECTION PROCEDURES USED

IP 37550	Onsite Engineering
IP 37551	Engineering
IP 51053	Electrical Components and Systems - Work Observation
IP 71111.08	Inservice Inspection Activities
IP 71111.17	Permanent Plant Modifications
IP 71111.23	Temporary Plant Modifications
IP 71152	Identification and Resolution of Problems
IP 92701	Follow-up
IP 50090	Pipe Support and Restraint Systems

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

259/2006-07-01	NCV	Failure to Construct Instrument Tubing Supports in Accordance with Design Drawings (Sections E1.8 and E1.9)

Closed

88-11	GL	Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations Mode (Section E8.1)
92-01	GL	Reactor Vessel Structural Integrity, Rev. 1 and Rev. 1 Supplement 1 (Section E8.2)
83-28	GL	Required Actions Based on Generic Implications of Salem ATWS Event (Section E8.3)
79-12	BUL	Short Period Scrams at BWR Facilities (Section E8.4)
88-07	BUL	Power Oscillation in Boiling Water Reactors (Section E8.5)
94-02	GL	Long-Term Solutions and Upgrade of Interim Operating Recommendations For Thermal Hydraulic Instabilities in BWRs (Section E8.6)
I.D.1	TMI	Control Room Design Review (Section E8.7)
I.D.2	TMI	Safety Parameter Display System (Section E8.8)
89-06	GL	Safety Parameter Display System (Section E8.8)
89-13	GL	Service Water System Problems Affecting Safety-Related Equipment (Section E8.9)
88-01	GL	NRC Position on IGSCC in BWR Austenitic Stainless Steel (Section E8.10)
05-09-02	IFI	Lower RVP Head Deposits (Section E8.11)
B-011	MPA	Flood of Equipment Important to Safety (Section E8.12)
B-041	MPA	Fire Protection - Final Technical Specifications (Section E8.13)
A-7	USI	Mark 1 Long Term Program, NuReg 0661, Supplement 1 (Section E8.14)
A-9	USI	ATWS (Section E8.15)

A-24	USI	Qualification of Class 1E Safety-Related Equipment (Section E8.16)
A-26	USI	Reactor Vessel Pressure Transient Protection (Section E8.17)
2515/95	TI	Inspection for Verification of BWR Recirc Pump Trip (Section E8.3)
C-002	MPA	Inspection for Verification of BWR Recirc Pump Trip (Section E8.3)
2500/20	TI	Inspection to Determine Compliance with ATWS Rule, 10 CFR 50.62 (Section E8.3)
A-075	USI	Salem ATWS Post-Trip Data Review and Information Capability (Section E8.3)
B-085	MPA	Salem ATWS Post-Trip Data Review and Information Capability (Section E8.3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section E1.1: Plant Modifications

Procedures and Standards

SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 9
MAI-4.2B, Piping, Rev. 20
G-94, Piping Installation, Modification, and Maintenance, Rev. 2
1-POI-64-2, MSIV Secondary Containment System, Rev. 0

DCNs

51106, Control Room Design Review (CRDR) - Control Bay1
51129, Appendix R and 240V Emergency Lighting - Reactor Building, Systems 999 and 247
51198,High Pressure Coolant Injection Mechanical - Reactor Building, System 73
51206, Control Rod Drive (CRD) Electrical and Mechanical - Reactor Building, System 85
51214, 120V AC Distribution Electrical - Reactor Building, System 57-2
51222,RHR Electrical - Reactor Building, System 74
61728, 48V DC Distribution - Control Bay, System 57-6

Work Orders

03-004725-16, core drilling
03-004725-17, core drilling
03-015067-03, battery pack emergency light replacement
03-015067-05, battery pack emergency light replacement
03-015067-09, battery pack emergency light replacement
03-015067-10, battery pack emergency light replacement
03-006734-22, system turnover punchlist items for setpoint and scaling issues

Section E1.2: Temporary Modifications

Procedures, Guidance Documents, and Manuals

0-TI-405, Plant Modifications and Design Change Control, Rev. 0
0-TI-410, Design Change Control, Rev. 1
SPP-9.5, Temporary Alterations, Rev. 6

TACFs

1-84-050-47, install a grab sample valve in the Unit 1 cooling water system for the main generator output breaker, 1-PCB-35-214.
1-85-002-85, Rev. 1, document pressure gauges in the CRD system
1-85-007-68, cut out the 3/4 inch vent line in the RCW system valve 1-FCV-68-03
1-85-030-68, document the existence of flanges and unions in the injector water relief piping of the CRD system
1-85-032-99, install jumpers in the RPS system to prevent scrams during the HFA relay coil change

1-85-016-99, remove fuses in the RPS system to inhibit the backup scram function during the HFA relay coil change out
0-04-004-023, RHRSW inlet bay supply line sluice gate closure

Misc Documents

Final Safety Analysis Report (FSAR) Rev. 20.09, Sections: 1.6.1.1.10 Design Bases Dependent Upon the Site and Environs, 2.42 Hydrology, 6.0 Core Standby Cooling Systems, 10.9 RHRSW System, App. F.7.7 Pumping Station (Intake Building), App. F.7.16 RHRSW
Calculation MDQ0023870123 Available NPSH for RHRSW Main Pumps, Rev. 7
Calculation ND-Q0023-970010 Suppression Pool Temperature Response for Non-Accident Unit Shutdown with Loss of Offsite Power, Rev. 3
Operating Instruction 0-OI-23 RHRSW System, Rev. 60
Technical Specifications 3.4.7 RHR SDC - Hot, 3.4.8 RHR SDC - Cold, 3.6.2.3 RHR Supp Pool Cooling, 3.6.2.4 RHR Supp Pool Spray, 3.6.2.5 RHR DW Spray, 3.7.1 RHRSW System, 3.7.2 EECW and UHS
General Design Criteria Document BFN-50-7023, Rev. 10
Abnormal Operating Instruction AOI 0-AOI-100-4 Breach of Wheeler Dam, Rev. 10

Section E1.3: System Return to Service Activities

Procedures, Guidance Documents, and Manuals

1-TI-437 System Return to Service (SRTS) Turnover Process for Unit 1 Restart, Rev. 16
General Engineering Specification G-40 Installation, Modification, and Maintenance of Electrical Conduit, Cable Trays, Boxes, Containment Electrical Penetrations, Electrical Conductor Seal Assemblies, Lighting, and Miscellaneous Systems, Rev. 15
Design Standard DS-E13.1.7 Dimensions of Rigid and Flexible Metal Conduit Bends
MAI-3.1 Installation of Electrical Conduit Systems and Conduit Boxes, Rev. 35

Problem Evaluation Reports (PERs)

105642 System 80 SPOC II Walkdown Bend Radius Issue

Miscellaneous Documents

Integration Task Equipment List (ITEL) System Scoping Milestone Reports for Systems 70, 80, and 57-4.
Calculation EDQ199920030014 Analysis of Flex Conduit to Devices for Unit 1, Rev. 2

Section E1.4: Area Turnover Program

Procedures and Standards

Business Practice BP-338, Area Turnovers from Recovery Unit 1 Restart Project, Rev. 1
DCN 51208 System 100 Penetrations and Sleeves, Rev. A
MAI 1.3 General Requirements for Modifications, Rev. 21
MAI 2.2 Mechanical Penetration Seals, Rev. 18
MAI 3.4A Internal Conduit Seals, Rev. 15

PER 104372 Debris and Oil on RB639
SPP 10.7 Housekeeping/Temporary Equipment Control, Rev. 1

Section E1.5: Restart Test Program

Procedures and Standards

Technical Instruction 1-TI-469, Baseline Test Requirements, Rev. 1
SSP-3.1, Corrective Action Program, Rev. 9
SPP-8.1, Conduct of Testing, Rev. 3
SPP-8.3, Post Modification Testing, Rev. 6
SSP-9.5, Temporary Alterations, Rev. 7
SSP-10.3, Verification Program, Rev. 1.

Restart Test Procedures

1-PMTI-BF-51090-STG53, Rev. 0
1-PMTI-BF-51102-STG02, Rev. 1 and Rev. 2
1-PMTI-BF- 51102-STG01, Rev. 1 and Rev. 2
1-PMTI-BF- 51018-STG03, Rev. 1
1-PMTI-BF- 51102-STG03, Rev. 0, STG04, Rev. 0, and STG15, Rev. 0
1-PMTI-BF-51177- STG05, Rev. 0
1-PMTI-BF- 51102-STG08, Rev. 0, and STG 10, Rev. 0
1-PMTI-BF- 51018-STG04, Rev. 1
1-PMTI-BFN- 51177- STG06, Rev. 1
1-PMTI-51368-STG04A, Rev. 0
1-PMTI-51368-STG04B, Rev. 0
1-PMTI-51368-STG04C, Rev. 0
1-PMTI-51368-STG04D, Rev. 0

Surveillance Instructions

0-SI-4.11.A.1(1), Local Fire Control Panel 0-LPNL-025-0555 Control Bay Elevation 593, Rev. 11.
0-SI-4.11.A.1(2), Local Fire Control Panel 0-LPNL-025-0556 Control Bay Elevation 606 Detection Operability Test, Rev. 06.
0-SI-4.11.A.1(3)A, Local Fire Control Panel 0-LPNL-025-0556 Control Bay Elevation 617 Detection Operability Test, Rev. 04.
0-SI-4.11.A.1(3)B, Local Fire Control Panel 0-LPNL-025-0556 Control Bay Elevation 617 Detection Operability Test, Rev. 04.
0-SI-4.11.A.2(1), Local Fire Control Panel 0-LPNL-025-0556 Control Bay Elevation 606 Miscellaneous Inputs Panel Test, Rev. 12.
0-SI-4.11.A.2(2), Supervised Fire System Circuit Operability Test, Rev. 19.
0-SI-4.11.A.1(4)B, Local Fire Control Panel 0-LPNL-025-0538, Intake Pumping Station and Cable Tunnel Detection Operability Test
0-SI-4.11.A.2(2), Supervised Fire System Circuit Operability Test, for Panel 0-LPNL-025-0538;
0-SI-4.11.B.1.E, Valve Cycling - High Pressure Fire Protection System
0-SI-4.11.B.1.C, System Flush - High Pressure Fire Protection System; and 0-SI-4.11.C.1.C, Simulated Automatic Actuation of the Fire Protection Sprinkler System.

<u>Section E1.6: Special Program Activities - Cable Installation and Cable Separation</u>

<u>Procedures and Standards</u>

BFN-50-728, DCD Physical Independence of Electrical Systems, Rev. 16
BFN-50-758, Power, Control, and Signal Cables for Use in Class 1 Structures, Rev. 15
BFN-50-7001, DCD Main Steam System, Rev. 19
DS-E12.1.5, Minimum Radius for Field-Installed Insulated Cables Rated 15KV and Less, Rev. 5
DS-E13.1.7, Dimensions of Rigid and Flexible Metal Conduit Bends, Rev. 3
DS-E13.6.2, Use of Rigid Conduit Bodies and Flexible Conduit Angle Connectors in Conduit Systems, Rev. 4
G-38, Installation, Modification and Maintenance of Insulated Cables Rated up to 15,000 Volts, Rev. 20
G-40, Installation, Modification and Maintenance of Electrical Conduit, Cable Trays, Boxes, Containment Electrical Penetrations, Electric Conductor Seal Assemblies, Lighting and Miscellaneous Systems, Rev. 15
MAI-3.2, Cable Pulling for Insulated Cables Rated up to 15KV Units 1, 2, and 3, Rev. 41
MAI-1.3, General Requirements for Modification, Rev. 21
NEDP-1, Design Basis and Design Input Control, Rev. 5
NEDP-3, Drawing Control, Rev. 10
NEDP-11, Design Input Walkdown Controls, Rev. 5
SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 13

<u>Work Order Packages</u>

W78030303005, Nuclear Plant-Request for Administrative Change to Drawing, Complete: 3/3/3

<u>Calculations</u>

EDQ199920030015, Analysis of Unit 1 Cable Installation - Miscellaneous Issues, Rev. 7
EDQ199920030061, Appendix G - Board/Panel Separation Analysis, Rev. 1
EDQ199920030061, Appendix A - PCIS Isolation Valve List, Rev. 2
EDQ199920030061, Appendix C - Control Board List, Rev. 0
EDQ199920030061, Appendix D - Control Board List, Rev. 0
EDQ199920030061, Appendix F - Control Board List, Rev. 0
EDQ0999910078, External Cable Separation Analysis, Rev. 8
EDQ0999910078, Attachment 235 External Cable Separation Analysis, Rev. 8
EDQ0000880437, Walkdown and Analysis of Panel Separation Violations, Rev. 3

<u>DCNs</u>

51016, Modify U1 RHR, Core Spray Initiation and Load Shed Logic
51194, Unit 1 Recovery Reactor Building Mechanical Lead System 069
51211, BFNP U1 Restart-Electrical Lead DCN-System 001
51223, BFNP U1 Restart-Electrical Lead DCN-System 075
51243, BFNP U1 Restart-Electrical Lead DCN-System 064, Stage 09
51090, BFNP U1 Restart-Electrical Lead DCN-System 57-4-480V Electrical Distribution (CB Bldg), Stage 73

Problem Evaluation Reports

99297, Cable Bend Radius Violation
100860, Cable Pullby Without Authorization
102329, Cable installation on improper side
101868, DCN 51090 Stages 83, 84 Separations
102757, Extent of Condition for PER 101868
105837, Conduit runs
104357, V3 and V4 cables installed in V3 penetration and tray

Post Issuance Changes (PIC)

64032B, Completed 10/27/05
63066, Issued 2/10/05
62657, Issued 2/22/05
63577, Issued 3/24/05

Drawings

1-45N1641-1, Wiring Diagrams Unit Control Boards Panel 9-3 Sh. 1, Rev. 1
1-45N1641-2, Wiring Diagrams Unit Control Boards Panel 9-3 Sh. 2, Rev. 3
1-45N1641-3, Wiring Diagrams Unit Control Boards Panel 9-3B Sh. 3, Rev. 4
1-45N1641-4, Wiring Diagrams Unit Control Boards Panel 9-3B Sh. 4, Rev. 3
1-45N1641-5, Wiring Diagrams Unit Control Boards Panel 9-3A Sh.5, Rev. 1
1-45N1641-6, Wiring Diagrams Unit Control Boards Panel 9-3A Sh. 6, Rev. 2
1-45N1641-7, Wiring Diagrams Unit Control Boards Panel 9-3A Sh. 7, Rev. E
1-45N1641-1, DCA 51090-602, Rev. R000
1-45N1641-4, DCA 51103-088, Rev. R2
1-45N1641-3, DCA 51103-138, Rev. C
1-45E1641-1A, DCA 51103-279, Rev. R000
1-45N1641-3, DCA 51094-845, Rev. RC
1-45N1641-3, DCA 51083-003, Rev. R2
1-45N1641-4, DCA 51094-667, Rev. R2
1-45N1641-5, DCA 51103-103, Rev. R001
1-45N1641-5, DCA 51083-014, Rev. R1
1-45N1641-5, DCA 51094-725, Rev. R001
1-45N1641-5, DCA 51094-726, Rev. R003
1-45N1641-5, DCA 51094-786, Rev. R1
1-45N1641-5, DCA 51094-843, Rev. R001
1-45N1641-6, DCA 51081-344, Rev. R2 CC
1-45N1641-6, DCA 51081-347, Rev. R2 CC
1-45N1641-7, DCA 51081-397, Rev. R2 CC
1-45N1641-7B, DCA 51081-334, Rev. R0 CC
1-45N1641-7B, DCA 51094-722, Rev. R000
1-45N1668, DCA 51081-156, Rev. R0 CC
1-45N1668, DCA 51081-136, Rev. R0 CC
1-45N1671-3, DCA 51081-158, Rev. R1 CC
45N1641-3, DCA 51076-068, Rev. 00
45N1641-3, DCA 51083-007, Rev. C
45N1641-3, DCA 51083-009, Rev. C
45N1641-3, DCA 51083-019, Rev. C

45N1641-3, DCA 51103-086, Rev. C
45N1641-3, DCA 51094-717, Rev. RC
45N1641-3, DCA 51094-785, Rev. RC
45N1641-3, DCA 51081-346, Rev. RC AC
45N1641-5, DCA 51081-185, Rev. R1 CC
45N1641-7, DCA 51081-397, Rev. RE AC
45N1641-7, DCA 51095-160, Rev. RE
45N1641-7, DCA 51094-668, Rev. E
45N1641-7, DCA 51094-720, Rev. RE
45N1641-7, DCA 51094-721, Rev. RE
45N1641-7, DCA 51103-202, Rev. E
45N1641-7B, DCA 51081-334, Rev. R0 CC
1-791E165-1, ARRGT Panel 9-3 Physical Drawing Rev. 38
1-730E933-1, DCA 51103-194, Rev. 3
1-791E165-2, ARRGT Panel 9-3 Physical Drawing Rev. 0
1-791E165-3, ARRGT Panel 9-3 Physical Drawing Rev. 0
1-791E343-3, DCA 51016-105, Rev. 0

Section E1.7: Special Program Activities - Q List

Procedures

SPP-9.3, Plant Modifications and Engineering Change Control
SPP-9.6, Master Equipment List, Rev 8
NEDP-4, Q-List and UNID Control, Rev 11
0-TI-414, Component Labling, Signs, operator Aids, and Permanent Information postings, Rev 3
MMDP-1, Maintenance Management System
SPP-9.6, Master Equipment List (MEL), Rev 8
NEDP-4, Q-List and UNID Control, Rev 11
0-TI-414, Component Labeling, Signs, Operator Aids, and Permanent Information Postings, Rev 3

NA Observation Reports

35104, Review of Q-List
37112, Review of MEL package processing
37290, RHR and RWCU MEL updates for retained components
38783, RHRSW MEL package updates

PERs

49089, Untimely loading of large MEL packages causing delays
53018, Lack of understanding of ownership MEL data update requirements
53020, Lack of understanding of ownership MEL data update requirements
53022, Inadequate communication of MEL project expectations
53023, Need to provide labling requirements in parallel with DCN issuance
53024, Very large MEL update packages issued rather than smaller packages
96459, Incorrect EMPAC status for EQ flow transmitter following MEL update
102475 Non-Qualified Beta Shielding Tape

Other Documents

Enterprise Maintenance and Planning Control (EMPAC) Database
Self Assessment, BFR-RSU-04-001, MEL Data

Section E1.8: Special Program Activities - Small Bore Piping and Instrument Tubing

Specifications & Procedures

TVA General Engineering Specification G-43, Installation, Modification, and Maintenance of Pipe Supports and Pipe Rupture Mitigative Devices, Rev. 13
TVA General Engineering Specification G-32, Bolt Anchors set in Hardened Concrete, Rev. 21
TVA General Engineering Specification G-29A, PS 0.C.1.2, Specification for Welding of Structures Fabricated in Accordance with AISC Requirements for Buildings and Inspected to the Criteria of NCIG-01
TVA General Engineering Specification G-29-S01, PS 4.M.4.4, ASME Section III and Non-ASME (Including AISC, ANSI B31.1 and ANSI B31.5)
Procedure No. N-VT-6, Visual Examination of Structural Welds Using the Criteria of NCIG-01, Rev 6
MAI-4.2A, TVA-BFNP Piping/Tubing Supports, Rev. 33

Drawings

Drawing number 0-47B435-1 through -21, Mechanical General Notes, Pipe Supports
Drawing numbers 1-47B452–2060-01 through -2060-2068, -2069, -2070, -2072, -2074, -2075, -2076, -3297 0-47B36-66, 1-47B456-2064-01 through - 2064-04, and 1-47B456-2064-07 through -18, and 147B456-2064-43, -2064-44, Mechanical RCIC System Pipe Support
Drawing numbers 1-47B600-5416-36, 1-47B600-5416-37, 1-47B600-5416-43, and 1-47B600-5438 through 1-47B600-5442, Mechanical Primary Containment System Pipe Support
Drawing number 1-47B466-31, Mechanical CRD System Pipe Support

Calculations

Calculation number CDQ1-071-2002-0824, Rev. 1, Small Bore Piping and Supports Program System calculation for the Unit 1 Seismic Class RCIC System (71) Piping

PERs

96004, Missing Support On System 71 Instrument Line
106420, Incorrect Unistrut Channel Installed on Support Number 1-47B600-5440

Miscellaneous Documents

TVA Nuclear Engineering Civil Design Standard DS-C1.7.1, General Anchorage to Concrete, Rev 9, dated 8/25/99
General Design Criteria Document BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing), Rev. 5, dated 9/9/91
General Design Criteria Document BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7, dated 4/6/94
General Design Standard DS-C1.2.6, General Pipe Support Design Manual, Rev. 0
Assessment Report BFN-REN-04-007, Small Bore Piping Program BFN Unit 1 Restart

DCN 51413, Install Small Bore Supports for RCIC System (System71) in Unit1 Reactor Building
DCN 51418, Install Small Bore Supports for Primary Containment System Isolation (System 64) in Unit1 Reactor Building
Engineering Change Control Documents, Post Issue Change (PIC) numbers 65455 and quality control inspection records documenting inspections pertaining to changes documented on PIC 65455, Work Order 03-014817-011

Section E1.9: Special Program Activities - Instrument Sensing Lines

Specifications & Procedures

TVA Engineering Specification N1E-003, Instrument and Instrument Line Installation and Inspection, Rev. 1
MAI-4.2A, TVA-BFNP Piping/Tubing Supports, Rev. 33
MAI-4.4A, TVA-BFNP Instrument Line Installation, Rev. 16
MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Rev. 4, dated 1/15/03
WI-BFN-1-GEN-01, General Requirements for Walkdowns, Rev. 4
WI-BFN-1-MEB-01, Walkdown Instruction for Mechanical and Instrumentation and Control Systems, Rev. 0

Drawings

Drawing number 0-47B435-1 through -21, Mechanical General Notes, Pipe Supports
Drawing numbers 0-47W600-97, Mechanical instrumentation and Controls
Drawing numbers 0-47E600-2560, Mechanical instrumentation and Controls, RHRSW Flow Detection HP Sensing Line for HX “B”
Drawing numbers 0-47E600-2561, Mechanical instrumentation and Controls, RHRSW Flow Detection LP Sensing Line for HX “B”
Drawing numbers 0-47E600-2564, Mechanical instrumentation and Controls, RHRSW Flow Detection HP Sensing Line for HX “D”
Drawing numbers 0-47E600-2565, Mechanical instrumentation and Controls, RHRSW Flow Detection LP Sensing Line for HX “D”
Drawing numbers 1-47B600-4887, through 4893, 4895, 4896, 4897, 4903 through 4907, 4909, 4910, 4912, 4913, 4915, and 5735, Mechanical RHRSW System, Pipe Support

PERs

94867, Installation of RVLIS Piping with Incorrect Slope
95171, Two Instrument Lines on System 68 Could not be Installed to Meet Slope Requirements
95175, During Installation of 1-LS-003-189, it was Determined that Support 1-47B415 was Installed Incorrectly to Maintain Slope
95316, Incorrect Slope on Portion of RHRSW Flow instrumentation
95432, Slope discrepancies on RCIC Instrument Line
95548, Incorrect Slope on Core Spray Cooling System Instrument Line Going to Panel 25-60
95734, Negative Slope on 4 Inch Section of Instrument Line
95836, Weld Removal Violated Minimum Wall Thickness of Instrument Line
96099, Discrepancies Between Design drawings and As-Found Conditions on System 68 Instrument Line
96115, Incorrect Slope on Portion of Instrument Line - System 68
96176, Incorrect Slope on Portion of Instrument Line - System 68
97029, System 85 Sensing Line Installed with Incorrect slope

98144, System 85 Sensing Line Installed with Incorrect slope
100956, System 74 Sensing Line Installed with Incorrect slope
102320, Incorrect Clamps on Support Numbers 1-47B600-4905 & 4910

102375, Inconsistencies in As-Built Drawing
102483, System 73 Sensing Line Installed with Incorrect slope
103893, Incorrect closure of PER 95316 by Field Engineering
104456, Incorrect Slope on Portion of Instrument Line - System 85
105842, Listing of Instrument sense lines which Perform Primary Safety Function
105959, Installation of Inlet and Outlet piping from RWCU Panel to Cooler in Reverse Order

Miscellaneous Documents

General Design Criteria Document BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing), Rev. 5, dated 9/9/91
General Design Criteria Document BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7, dated 4/6/94
General Design Standard DS-C1.2.6, General Pipe Support Design Manual, Rev. 0
Walkdown Package WDP-BFN-1-EEB-23-04-INST
Sample calculation - Instrument Sensing Lines , BFR Unit 1 Slope, Separation and Materials Evaluations
TVA Nuclear Engineering Civil Design Standard DS-C1.7.1, General Anchorage to Concrete, Rev 9, dated 8/25/99
DCN 51177, Modifications to Instrument Line Slope, and Exception Number EX-NE1E-003-57
Work Order numbers 03-019416-002 and 03-019416-003 and quality control inspection records for RHR service water instrument sensing lines (System 23)

Section E1.10: Special Program Activities - Large Bore Piping and Supports

Procedures and Design Criteria

Procedure No., WI-BFN-0-CEB-01, Walkdown Instruction for Piping and Pipe Supports
Design Criteria BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7
MMDP-10, Rev. 6, Controlling Welding, Brazing, and Soldering Processes

Other Documents

Pipe Support Drawing No. 1- 47B406-285, Rev. R003
Pipe Support Drawing No. 1- 47B406-287, Sheets 1 & 2, Rev. R002 & 005
Pipe Support Drawing No. 1- 47B406-290, Rev. R001
Pipe Support Drawing No. 1- 47B452-3037, Rev. R003
Pipe Support Drawing No. 1- 47B465-464, Rev. R002
Pipe Support Drawing No. 1- 47B465-500, Sheets 1 & 2, Rev. R004 & 002
Pipe Support Drawing No. 1- 47B465-501, Rev. R001
Pipe Support Drawing No. 1- 47B465-507, Rev. R002
Pipe Support Drawing No. 1- 47B465-546, Sheets 1 & 2, Rev. R004 & 002
Support Calculation CDQ1-068-2002-0216, Rev. 003 for Support 1-47B465-546
Bechtel Corp. Computer Program, Version 19, Frame Analysis Program for Pipe Supports (FAPPS)
Bechtel Computer Program ME 153, Rev. 10, Miscellaneous Application Program for Pipe Supports (MAPPS)

Bechtel Computer Program ME 035, Rev. 12, Baseplate
Isometric Drawings for Piping Stress Problem N1-123-2R, Sheets 6 & 7 & Related Pipe Support As-Built Drawings (System 23)
Isometric Drawings for Piping Stress Problem N1-178-1R Sheet 7 & N1-178-2RF, Sheet 9 & Related Pipe Support As-Built Drawings (System 78)
Drawing Title: Unit 1 Seismic Class 1 Boundary Residual Heat Removal System for Calculation No. ND-Q0999-920011, Rev. 42, Seismic Class 1 System Piping Boundary
Isometric drawings (partial) for Stress Problems N1-174-1RA, -2R to -5R, -7R, -9R, -11R, -14R, -15R, -17R, -18R, -27R, and 28R
DCN 51028, core Spray system Large Bore for Bulletins 70-02/79-14
DCN 51347, residual Heat Removal System for Bulletins 79-02/79-14
Work Order (WO) 02-009379-001 for DCN 51028
WO 03-008315-000 for DCN 51347
Technical Instruction 0-TI-449, NDE Matrix/Welding Engineer Guide, Rev. 1
TVA 79-14 Design/Configuration Control Process for Unit 1 vs. Unit 2/3D
PER 104573, Detail B3037 Was Cut from the Wrong Item Member
PER 104574, Pin Diameter for Item 7 Was Accidently Changed by the Drafter

Section E1.11: Special Program Activities - Fuse Program

Fuses in the Following MCC Cubicles Were Inspected
480 V RMOV BD 1A Cubicles 4D, 5A, 17E, 16A
480 V RMOV BD 1B Cubicles 13C3-7A, 7B
250 V RMOV BD 1A Cubicles R1A, 8B2
250 V RMOV BD 1B Cubicle 5B
250 V RMOV BD 1C Cubicle 1E

Calculations Reviewed

ED-Q0248-920318, Modifications to 250 V DC Shutdown Boards A, B, C, D and 3EB Fuse Evaluation, Rev. 14
ED Q0009880524, Evaluation of DC Fuses in Miscellaneous Panels, Rev. 30
ED-Q0248-880141, Fuse Program - 250 V DC Shutdown Boards A/B/C/D/3EB, Rev. 006
ED-Q0211-880138, Fuse Evaluation for 4KV Shutdown Boards A, B, C, D, 3EA, 3EB, 3EC, 3ED, Rev. 015
ED-Q021-92070, Modifications to Fuse Evaluation for 4 KV Shutdown BDs A, B, C, D, 3EA, 3EB, 3EC, 3ED, Rev. 018
ED-Q0280-880140, Fuse Program 250 V DC Battery Boards 1-2-3, Rev. 015
ED-Q0231-880133, Fuse Program 480 V Shutdown Boards 1A/B, 2A/B, 3A/B, Rev. 021
ED-Q0231-920310, Mini-Calculation for Modifications to Fuse Evaluation for 480 V SD Boards 1A/1B/2A/2B/3A/3B, HVAC BD B, Div. I & II Load Shed Logic Panels
EDQ0268880134, Fuse Program - 480 V Reactor MOV Boards 1A/B, Rev. 012
EDQ1-2B1-2002-0041, 250 V DC Reactor MOV BDs 1A, 1B, and 1C Fuse Evaluation, Rev. 005
EDQ0009880523, Fuse Program Panel 9-5, 9-9, 9-14, 9-15, 9-17, 9-42 and 9-43, Rev. 029
ED-Q0009-920497, Mini-Calculation for the Modification to the Fuse Program for Various Panels, Rev. 057

PERs Reviewed

74070
105757
105179
105195
104809

Section E1.12: Special Program Activities -Cable Splices

Procedures

MAI - 3.3, Cable Terminating and Splicing for Cables Rated Up to 15,000 Volts, Rev. 50
PII-57083, Rev. AB, Raychem Product Installation Instructions for NMCK8-L Kits
PII-57147-B, Raychem Nuclear Plant Kit Installation Instructions - Transition Splice

PERs

98178
98052
95080
94470
98171
98174
98176
106776
104256

Completed Work Orders

02-015487-033
02-015487-067
02-015981-051
03-006607-027

Section E1.13: Special Program Activities - Thermal overloads

TOLs and End Devices inspected

Components	MCCs	Cubicles	TOLs
1-FCV-023-0034	480 V RMOV BD 1A	4D	strapped out
1-FCV-071-0003	250 V RMOV BD 1B	5B	C220A
1-FCV-073-0002	480 V RMOV BD 1A	17E	C125B
1-FCV-074-0047	250 V RMOV BD 1A	R1A	C180B
1-FCV-075-0022	480 V RMOV BD 1A	16A	C778A
1-MTR-73-10	250 V RMOV BD 1A	8B2	C184A
1-MTR-71-29	250 V RMOV BD 1C	1E	C125B
1-MTR-75-76	480 V RMOV BD 1B	13C3-7A	C867A
1-MTR-64-69	480 V RMOV BD 1B	7B	C778A
1-MTR-77-14A	480 V RMOV BD 1A	5A	C526A

Calculations

EDQ1-999-2002-0075, TOL Heater Calculation - Motor Operated Valves, Rev. 010
EDQ1-999-2002-0076, TOL Heater Calculation - Continuous Duty Motors, Rev. 004

Procedures

NEDP-8, Technical Evaluation for Procurement of Materials and Services, Rev. 12
SPP-9.3, Plant Modifcations and Engineering Change Control, Rev. 13

PERs

89577
106791
106704

Section E1.14: Special Program Activities - Moderate Energy Line Break

TVA letter to NRC dated January 23, 1991, “Nureg-1232, Volume 3, Supplement 2, Browns Ferry Unit 2 “Safety Evaluation Report on Tennessee Valley Authority: Browns Ferry Nuclear Performance Plan” for Unit 2.”
Report, W87 040615 001, Moderate Energy Line Break (MELB) Flood Evaluation Report for Browns Ferry Nuclear Plant Unit 1 Extended Power Uprate, Rev. 1, dated June 11, 2004.
NRC letter to TVA dated October 24, 1988, BFN Performance Plan.
TVA letter to NRC dated June 25, 2004, Completion of MELB Flooding Evaluation
NRC Letter to TVA dated June 7, 2006, Completion of Special Program - Moderate Energy Line Break Flooding Evaluation (TAC No. MC3689)

Section E1.15: Emergency Equipment Cooling Water Flow Testing

Procedures

0-TI-54, EECW System Operational Flush, Rev. 8
0-TI-63, RHRSW Flow Blockage Monitoring, Rev. 22
0-TI-154, Coupons and Monitoring for Corrosion and Deposit Control, Rev. 8
0-TI-389, Raw Water Fouling and Corrosion Control, Rev. 9
0-TI-522, Program for Implementing NRC Generic Letter 89-13, Rev. 0
1/2/3-SI-3.2.4, EECW Check Valve Test, Rev. 29, 36, 27
CHTP-108, Technical Chemistry Standards for SPP-9.7, Rev. 1
CI-137, Raw Water Chemical Treatment, Rev. 17
CI-137.5, Raw Water Chemical Treatment Molluscicide Control, Rev. 26
MCI-0-074-HEX001, Maintenance of RHR Heat Exchangers, Rev. 17
MCI-0-082-CLR001, Standby Diesel Engine Water Coolers Disassembly, Inspection, Rework and Reassembly, Rev. 27
SPP-9.7, Corrosion Control Program, Rev.12

Completed Work Orders

98-002712-000, Inspect/Clean RHRSW Pump Pit, 03/98
01-005424-000, Inspect/Clean RHRSW Pump Pit, 05/03

Other Documents

Project Plan, TVAN Raw Water Corrosion Program, Rev. A
System Health Reports, Residual Heat Removal Service Water and Emergency Equipment Cooling Water, 2005-2006
EECW Component Minimum Flow Trends, From 1/2/3-SI-3.2.4 Check Valve Test Procedure, 1995-2006

Section E1.16 : Inservice/Preservice Inspection

Procedures and Standards

1-SI-4.6.G, Inservice Inspection Program Unit 1, Rev. 5
SPP-9.1, ASME Section XI, Rev. 4
N-GP-31, Calculation of ASME Code Coverage for Section XI, Appendix VIII UT examinations, Rev. 0
N-UT-64, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, Rev. 8
N-PT-9, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, Rev. 28
N-MT-6, Magnetic Particle examination for ASME and ANSI Code Components and Welds, Rev. 27
N-VT-1, Visual Examination Procedure for ASME Section XI Preservice and Inservice, Rev. 38

PERs

105583*
105691*
105692*
105987*
88166
96535
94161
103414
102325
101545
97951
96646
96567
*PERs initiated as a result of NRC inspection

Section E8.3: Closeout of Generic Letter 83-28, Required Actions Based on Generic Implications of Salem ATWS Events

NRC Safety Evaluation for Generic Letter 83-28, Item 1.2, Post Trip Review (Data and Information Capability), dated June 12, 1985
NRC Safety Evaluation for Generic Letter 83-28, Item 1.1, Post Trip Review (Programs and Procedures)
NRC Safety Evaluation for Generic Letter 83-28, Item 2.1 (Part 1) Equipment Classification, dated September 2, 1986

NRC Safety Evaluation for Generic Letter 83-28, Items 3.1.3 and 3.2.3, Post-Maintenance Testing (RTS Components, All Other Safety-Related Components), dated October 27, 1986
NRC Safety Evaluation for Generic Letter 83-28, Item 2.1 (Part 2), Vendor Interface Program, dated May 1, 1989
NRC Safety Evaluation for Generic Letter 83-28, Item 2.2 (Part 1), Equipment Classification Program, dated June 1, 1989
NRC Safety Evaluation for Generic Letter 83-28, Item 4.5.3, Reactor Trip Reliability-On-line Functional Testing of the Reactor Trip System, dated August 17, 1990

Section E8.4: IEB 79-12 Short Period Scrams at BWR Facilities

Procedures

2-GOI-100-1A, Unit Startup and Power Operation , Rev. 107

Other Documents
Unit 1 Technical Specifications
NRC IE Bulletin 79-12 Short Period Scrams at BWR Facilities
TVA letter to NRC dated January 4, 1990, Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Office of Inspection and Enforcement Bulletin 79-12 - Commitment Revision Regarding Fast Period Scrams
NRC Inspection Report 50-259,260,296/95-51

Section E 8.5: IEB 88-07 and Supplement 1 Power Oscillation in BWRs

Procedures

2-GOI-100-1A, Unit Startup and Power Operation, Rev. 107
2-SR-3.3.1.1.I, Core Thermal Hydraulic Stability, Rev. 9
2-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable, Rev. 004
2-AOI-68-1B, Recirc Pump Trip/Core Flow Decrease, Rev. 003

Other Documents

Unit 1 Technical Specifications
NRC IEB 88-07 Power Oscillations in BWRs
NRC IEB 88-07 Supplement 1 Power Oscillations in BWRs
Problem Evaluation Report 102236 THI Manual Operator Actions
TVA letter to NRC dated November 4, 1988, Browns Ferry Nuclear Plant (BFN) - NRC Bulletin 88-07; Power Oscillations in BWRs
TVA letter to NRC dated March 6, 1989, Browns Ferry Nuclear Plant (BFN) - NRC Bulletin 8-07 Supplement 1; Power Oscillations in BWRs
TVA letter to NRC dated January 14, 1992, Browns Ferry Nuclear Plant (BFN) - TVA BFN Technical Specifications (TS) No. 300 Reactor Core Thermal-Hydraulic Stability
NRC letter to TVA dated May 31, 1994, Issuance of Technical Specification Amendment For the Browns Ferry Nuclear Plant Units 1 and 3 (TAC Nos. M82650 and M82652) (TS-300)
Simulator Exercise Guide HLTS-10, Rev. 0
TVA letter to NRC dated November 10, 2003 TVA-BFN-TS-430, Browns Ferry Nuclear Plant (BFN) Unit 1 - Technical Specification (TS) Change 430 - Power Range Neutron Monitor Upgrade With Implementation of Average Power Range Monitor and Rod Block Monitor Technical Specification Improvements and Maximum Extended Load Line Limit Analysis

TVA letter to NRC dated January 6, 2006, TVA-BFN-TS-443 Unit 1 - Technical Specification (TS) Change TS-443 - Oscillation Power Range Monitor

Section E8.6: GL 94-02 Long-Term Solutions and Upgrade of Interim Operating Recommendations For Thermal-Hydraulic Instabilities in BWRs

Procedures

2-GOI-100-1A, Unit Startup and Power Operation, Rev. 107
2-SR-3.3.1.1.I, Core Thermal Hydraulic Stability, Rev. 9
2-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable, Rev. 004
2-AOI-68-1B, Recirc Pump Trip/Core Flow Decrease, Rev. 003

Other Documents

Unit 1 Technical Specifications
NRC GL 94-02 Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal Hydraulic Instabilities in BWRs
TVA letter to NRC dated September 8, 1994, Browns Ferry Nuclear Plant (BFN) - Units 1,2 and 3 - Response To NRC Generic Letter (GL) 94-02 - Long-Term Solutions and Upgrade of Interim Operating Recommendations For Thermal-Hydraulic Instabilities in BWRs
BWR Owners Group letter to NRC dated May 19, 1995, Stability Long-Term Solution ABB Option III OPRM Generic Topical Report
TVA letter to NRC dated October 4, 1995, Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3 - TVA Confirmation of the Installation Schedule for the Stability Long-Term Solution for NRC Generic Letter (GL) 94-02
NRC Inspection Report 50-259, 260, 296/95-62
TVA letter to NRC dated May 24, 1999, Browns Ferry Nuclear Plant (BFN) - Unit 2 - Implementation of Long-Term Stability Solution - Option III
TVA letter to NRC dated May 25, 2000, Browns Ferry Nuclear Plant (BFN) - Unit 3 - Completion of Actions Associated With Generic Letter 94-02 For Implementation of Long-Term Stability Solution Option III
TVA letter to NRC dated July 25, 2001, Browns Ferry Nuclear Plant (BFN) Units 2 and 3 - Technical Specifications (TS) Change 415 - Deletion of 120-day Required Action for Restoration of Oscillation Power Range Monitor (OPRM) Function - Emergency TS Change Request for Unit 2
TVA letter to NRC dated December 31, 2003, Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Operational Power Range Monitors (OPRMs) - Plan to Support Operability

Section E8.7 :TMI I.D.1 Control Room Design Review

NRC Letter dated 8/9/88, Safety Evaluation for the Detailed Control Room Design, Browns Ferry Nuclear Plant, Units 1, 2, and 3
TVA Letter dated 1/9/91, Browns Ferry Nuclear Plant (BFN) – Plans for the Return to Service of BFN Units 1 and 3
TVA Letter dated 12/15/93, Browns Ferry Nuclear Plant – Units 1, 2, and 3 – Operating & Maintenance Cost Reduction Program – Revision of Detailed Control Room Design Review Program to Discontinue Cost Benefit Analysis of Non-Safety Significant Human Engineering Discrepancies (HEDs)

Generic Letter 82-33, Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability, dated 12/17/82
TVA Letter dated 8/22/91, Browns Ferry Nuclear Plant – Supplemental Response to NRC Safety Evaluation for the BFN Detailed Control Room Design Review
NRC Letter dated 10/29/91, Safety Evaluation for the Browns Ferry Nuclear Plant Detailed Control Room Design Review
TVA Letter dated 6/14/93, Browns Ferry Nuclear Plant – Completion of NUREG-0737 (TMI Action Plan), Item I.D.1, Control Room Design Reviews for Unit 2
TVA Letter dated 2/9/96, Browns Ferry Nuclear Plant – Completion of NUREG-0737 (TMI Action Plan), Item I.D.1, Control Room Design Reviews for Unit 3

Section E8.8: TMI I.D.2, Safety Parameter Display System and GL 89-06, Safety Parameter Display System

Miscellaneous Documents

DCN 51082, BFNP Unit 1 Recovery – I&C Lead DCN – System 261
BFN U1 ICS External Interface List
Generic Letter 89-06, Task Action Plan Item I.D.2 – Safety Parameters Display System- 10 CFR 50.54(f), dated 4/12/89
TVA Letter dated 1/14/85, Implementation Schedule for SPDS
TVA Letter dated 4/8/87, Browns Ferry Nuclear Plant – Safety Parameter Display System (SPDS) – Schedule for Response to Request for Additional Information
TVA Letter dated 9/19/88, Browns Ferry Nuclear Plant – Safety Parameter Display System (SPDS) – Response to Request for Additional Information
TVA Letter dated 12/19/89, Browns Ferry Nuclear Plant – Phase 1 Safety Parameter Display System (ISPDS)
TVA Letter dated 10/22/90, Browns Ferry Nuclear Plant – Notification of Implementation of NUREG-0737 (TMI Action Plan), Item I.D.2.1, SPDS, Phase 1 Installation and Final Design Description
TVA Letter dated 12/11/90, Browns Ferry Nuclear Plant – Notification of Implementation of NUREG-0737 (TMI Action Plan), Item I.D.2.1, SPDS, Final Design Description
NRC Letter dated 3/6/91, Interim and Final Design of the Safety Parameter Display System at the Browns Ferry Nuclear Plant
TVA Letter dated 12/17/91, Browns Ferry Nuclear Plant – Safety Parameter Display System, Response to NRC Safety Evaluation Report (SER) Open Items
NRC Letter dated 2/5/92, Safety Parameter Display System - Browns Ferry Nuclear Plant, Units 1, 2, and 3
TVA Letter dated 10/19/93, Browns Ferry Nuclear Plant – Completion of NUREG-0737 (TMI Action Plan), Item I.D.2.1, Plant Safety Parameter Display Console, and Certification in Accordance with Generic Letter 89-06

Section E8.9: GL 89-13, Service Water System Problems Affecting Safety-Related Equipment

Procedures

0-TI-54, EECW System Operational Flush, Rev. 8
0-TI-154, Coupons and Monitoring for Corrosion and Deposit Control, Rev. 8
0-TI-389, Raw Water Fouling and Corrosion Control, Rev. 9

0-TI-522, Program for Implementing NRC Generic Letter 89-13, Rev. 0
1-SI-3.2.4, EECW Check Valve Test, Rev. 29, 36, 27
CHTP-108, Technical Chemistry Standards for SPP-9.7, Rev. 1
CI-137, Raw Water Chemical Treatment, Rev. 17
CI-137.5, Raw Water Chemical Treatment Molluscicide Control, Rev. 26
MCI-0-082-CLR001, Standby Diesel Engine Water Coolers Disassembly, Inspection, Rework and Reassembly, Rev. 27
SPP-9.7, Corrosion Control Program, Rev.12

Other Documents

Corrosion Monitoring Coupon Trends, EECW, 1993-2006
EECW Component Minimum Flow Trends, From 1-SI-3.2.4 Check Valve Test Procedure, 1995-2006
Work Order PM Ticket History, Unit 1 EECW Heat Exchangers Cleaning and Inspection

Section M1: Conduct of Maintenance

Procedures and Standards

SPP-10.2, Clearance Program, Rev. 6
TI-106, General Leak Rate Test Procedure, Rev. 10

Work Orders

03-001414-39, support of the pneumatic testing of Drywell and Emergency Control Air
03-006734-29, replacement of the packing for bypass valve 1-BYV-85-551
03-019008-00, disassembly and inspection of flow control valve 1-FCV-84-19
03-019010-00, disassembly and inspection of flow control valve 1-FCV-84-20
03-017668-01, installation and testing of the new valve operator for flow control valve 1-FCV-85-83A
04-724879-12, removal and replacement of the 1C RHR Heat Exchanger floating head
06-710104-00, disassembly and inspection of check valve 1-CKV-32-655